

September 29, 2003

Mr. Gregg R. Overbeck  
Senior Vice President, Nuclear  
Arizona Public Service Company  
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Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNIT 2 (PVNGS-2) -  
ISSUANCE OF AMENDMENT ON REPLACEMENT OF STEAM  
GENERATORS AND UPRATED POWER OPERATIONS (TAC NO. MB3696)

Dear Mr. Overbeck:

The Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. NPF-51 for the Palo Verde Nuclear Generating Station, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 21, 2001, as supplemented by letters dated March 13, August 27, August 29, September 4, September 6, October 11, November 21, December 10, December 23, 2002, and March 11, June 10, July 25, and August 22, 2003.

The amendment changes the operating license and TSs to support replacement of the steam generators and subsequent operation at an increased maximum power level of 3990 MWt, a 2.94 percent increase from the current 3876 MWt. The amendment shall be implemented prior to entry into Mode 4 during the restart from the Fall 2003 refueling outage.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

**/RA/**

Bo M. Pham, Project Manager, Section 2  
Project Directorate IV  
Division of Licensing Project Management  
Office of Nuclear Reactor Regulation

Docket No. STN 50-529

Enclosures: 1. Amendment No. 149 to NPF-51  
2. Safety Evaluation

cc w/encls: See next page

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Senior Vice President, Nuclear  
Arizona Public Service Company  
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Sincerely,

/RA/

Bo Pham, Project Manager, Section 2  
Project Directorate IV  
Division of Licensing Project Management  
Office of Nuclear Reactor Regulation

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Docket No. STN 50-529

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2. Safety Evaluation

cc w/encls: See next page

\* See memo signed by person on that date

\*\* See previous concurrence

ADAMS Accession No.: 032720538 NRR-058 TS: ML032730666 NRR-100 PKG: ML032731029

OFFICE	PDIV-2/PM	PDIV-2/PM	PDIV-1/LA	OGC NLO w/comments	PDIV-2/SC
NAME	JDonohew	BPham	MMcAllister	RMWeisman	SDembek
DATE	8/28/2003	8/28/03	9/26/03	26 Sept 2003	9/29/03
	*EMEB/SC	*EMCB/SC	*EMCB/SC	*SPLB/SC	*SPSB/SC
	KManoly	LLund	SCoffin	SWeerakkody	MReinhart
	4/9/03	9/23/02	9/25/02	11/26/02	2/10/03
	*EEIB/SC	*SRXB/SC	**IEHB/SC	PDIV/D	DLPM/D
	CHolden	FAkstulewicz	REckenrode for DTrimble	HBerkow	LMarsh
	12/23/02	4/8/03 9/2/03	08/05/2003	9/29/03	9/29/03

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ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149  
License No. NPF-51

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority dated December 21, 2001, as supplemented by letters dated March 13, August 27, August 29, September 4, September 6, October 11, November 21, December 10, December 23, 2002, and March 11, June 10, July 25, and August 22, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-51 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 149, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented before entry into Mode 4 during the restart from refueling outage 11.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

Ledyard B. Marsh, Director  
Division of Licensing Project Management  
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and Technical  
Specifications

Date of Issuance: September 29, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 149

FACILITY OPERATING LICENSE NO. NPF-51

DOCKET NO. STN 50-529

Replace the following pages of the Facility Operating License No. NPF-51 and Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE

Operating License  
Page 5

Technical Specifications  
Page 1.1-6  
Page 3.3.1.8  
Page 3.3.2-5  
Page 3.3.5-4  
Page 3.4.1-3  
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Page 3.7.1-3  
Page 5.5-23

INSERT

Operating License  
Page 5

Technical Specifications  
Page 1.1-6  
Page 3.3.1.8  
Page 3.3.2-5  
Page 3.3.5-4  
Page 3.4.1-3  
Page 3.4.1-4  
Page 3.7.1-3  
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE

NO. NPF-51

POWER UPRATE AND STEAM GENERATOR REPLACEMENT AMENDMENT

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT 2 (PVNGS-2)

DOCKET NO. STN 50-529

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE

NO. NPF-51

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT 2 (PVNGS-2)

DOCKET NO. STN 50-529

**1.0 INTRODUCTION**

By application dated December 21, 2001, as supplemented by letters dated March 13, August 27, August 29, September 4, September 6, October 11, November 21, December 10, December 23, 2002, and March 11, June 10, July 25, and August 22, 2003, Arizona Public Service Company (APS or the licensee) requested to amend Facility Operating License NPF-51 and Appendix A, Technical Specifications for PVNGS-2. The proposed changes support replacement of the steam generators (SGs) and subsequent operation at an increased maximum power level of 3990 Megawatts thermal (MWt), a 2.94 percent increase from the current maximum power of 3876 MWt. This proposed power uprate (PUR) would be implemented during plant start-up after the SGs are replaced in PVNGS-2 during refueling outage No. 11, scheduled to begin in September 2003. Some of the proposed changes in the licensee's submittal are being made to accomplish the PUR, while others are needed both to accomplish the PUR and replacement steam generators (RSGs).

Attachment 6 of APS's December 21, 2001 amendment request, "Power Uprate Licensing Report (PURLR)," includes a reanalysis or evaluation by the licensee of the Large Break Loss of Coolant Accident (LBLOCA), Small Break Loss of Coolant Accidents (SBLOCAs), non-LOCA accidents, and Nuclear Steam Supply System (NSSS) and Balance-of-Plant (BOP) Structures, Systems and Components (SSCs). Major NSSS components (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps, and SGs) and BOP components (e.g., auxiliary feedwater, shutdown cooling, electrical distribution, emergency diesel generators, containment cooling and the ultimate heat sink) have been assessed with respect to the bounding conditions expected for operation at the PUR level. Control systems (e.g., reactor regulating, pressurizer pressure and level, turbine control, FW control and steam bypass control) have also been evaluated for operation at PUR conditions.

This Safety Evaluation (SE) documents the NRC staff's evaluation of the licensee's analyses mentioned above. In its review, the staff sought to determine whether or not the licensee's results are acceptable and demonstrate that the applicable design basis acceptance criteria will continue to be met during the PUR conditions with the RSGs. This SE contains reviews by NRC staff members from various technical disciplines, and is arranged by subject matter similar

to those outlined in the licensee's PURLR submittal. The Table of Contents lists all specific topics and subtopics discussed.

## 1.1 Description of Proposed Amendment

The proposed amendment would allow operation of Unit 2 up to a maximum reactor core power level of 3990 MWt, an increase of 2.94 percent above the current licensed power level of 3876 MWt. Specifically, the following Unit 2 Facility Operating License and Technical Specification (TS) changes are requested to support the increased power operation:

### Facility Operating License No. NPF-51

Revise paragraph 2.C.(1) to increase the authorized 100 percent reactor core power (rated thermal power) from 3876 MWt to 3990 MWt, an increase of 2.94 percent. The new power level of 3990 MWt represents an increase of 5 percent above the originally licensed power level of 3800 MWt. The increase to 3876 MWt was authorized by the NRC in a letter dated May 23, 1996, Amendment No. 100 for Unit 2.

### TS 1.1 Definitions

Revise definition of Rated Thermal Power for Unit 2, from 3876 MWt to 3990 MWt.

### TS 3.3.1 Reactor Protective System (RPS) Instrumentation - Operating

Revise Table 3.3.1-1, Reactor Protective System Instrumentation, Item 6, Steam Generator #1 Pressure - Low and Item 7, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Unit 2. This increase in the Allowable Value is proportional to the increase in SG pressure during normal operation and will ensure a comparable RPS response. Both the PUR and the RSGs affect this specification.

### TS 3.3.2 RPS Instrumentation - Shutdown

Revise Table 3.3.2-1, Reactor Protective System Instrumentation - Shutdown, Item 2, Steam Generator #1 Pressure - Low and Item 3, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Unit 2. This increase in the Allowable Value is proportional to the increase in SG pressure during normal operation and will ensure a comparable RPS response. Both the PUR and the RSGs affect this specification.

### TS 3.3.5 Engineered Safety Features Actuation System (ESFAS) Instrumentation

Revise Table 3.3.5-1, Engineered Safety Features Actuation System Instrumentation, Item 4.a, Steam Generator #1 Pressure - Low and Item 4.b, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Unit 2. This increase in the Allowable Value is proportional to the increase in SG pressure during normal operation and will ensure a comparable engineered safety features system response. Both the PUR and the RSGs affect this specification.

### TS 3.4.1 Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

Revise Figure 3.4.1-1, Reactor Coolant Cold Leg Temperature vs. Core Power Level, to change the upper limit in the area of acceptable operation for Unit 2. The new upper limit line would allow a cold leg temperature of 570°F at 0 percent power, decreasing linearly to 564°F at 100 percent power. The upper limit line of Figure 3.4.1-1 in the current TS decreases linearly from 570°F at 0 percent power to 568°F at 30 percent power. At 30 percent power the current figure then decreases linearly from 568°F to 560°F at 100 percent power. The increase in cold

leg temperature at 100 percent power will allow a more optimum main steam pressure for turbine operation. Both the PUR and RSGs affect this specification.

#### TS 3.7.1 Main Steam Safety Valves (MSSVs)

Revise Table 3.7.1-1, Variable Overpower Trip (VOPT) Setpoint versus OPERABLE Main Steam Safety Valves for Unit 2, to decrease the Maximum Power and the Maximum Allowable VOPT Setpoint when the Minimum Number of MSSVs per Steam Generator Required Operable is less than 10. The reduction in allowable power levels and VOPT setpoints for Unit 2 are needed to offset the impacts of increased core power and increased cold leg temperature on the consequences of the Updated Final Safety Analysis Report (UFSAR) Chapter 15 design basis events. The PUR affects this specification.

#### TS 5.5.16 Containment Leakage Rate Testing Program

Revised to increase the peak calculated containment internal pressure for the design basis loss of coolant accident ( $P_a$ ) for Unit 2 from 52.0 psig to 58.0 psig. The proposed value for  $P_a$  has been rounded up from the actual calculated value of 57.85 psig. The calculated peak containment pressure remains below the containment internal design pressure of 60.0 psig. Both the PUR and the RSGs affect this specification.

#### Bases for Technical Specifications 3.6.1, 3.6.2, 3.6.4, 3.6.6 and 3.7.1

Bases would be revised to reflect these changes described above.

## **2.0 NUCLEAR STEAM SUPPLY SYSTEM**

### **2.1 Emergency Core Cooling System**

The emergency core cooling system (ECCS) is designed to provide core cooling in the event of a LOCA. The objectives of the ECCS are: to maintain the core subcritical, to remove decay heat in order to maintain core coolable geometry, limit cladding water interaction, prevent fuel melting, and remove core decay heat for an extended period of time. In PVNGS-2, the ECCS consists of two high-pressure safety injection (HPSI) systems, two low-pressure safety injection (LPSI) systems and four safety injection tanks (SIT). The HPSI and LPSI are arranged as two active redundant trains. Each SIT injects into a cold leg and in the case of blowdown will provide borated cooling water until the other two (active) systems are energized.

The staff reviewed the licensee's submittal to verify that the ECCS system is able to adequately perform its function for the PUR. As shown in the analyses and results of Section 4.1 of this SE, "Emergency Core Cooling System Performance Analysis," the staff concludes that the licensee's ECCS system is of appropriate size and capacity to protect the reactor core during a LOCA event. Therefore, the ECCS system is acceptable at the PUR condition.

### **2.2 Containment Heat Removal System**

The licensee uses the COPATTA containment analysis code to calculate the containment response to a LOCA or main steamline break (MSLB). COPATTA assumes a spray drop size larger (and therefore more conservative) than the actual drop size produced by PVNGS-2 containment spray system (CSS) nozzles. The drop size is affected by the containment pressure. The PUR results in a higher containment pressure. However, the drop size

assumption in COPATTA remains bounding. The spray distribution and spray flow rate assumed in the analysis are bounding and therefore are unaffected by the PUR. For these reasons the staff finds the PUR's effects on the CSS to be acceptable.

### **3.0 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS**

This section of the staff's review focuses on verifying the adequacy of the structural and functional integrity of piping systems, components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes. The acceptance criteria are based on continued conformance with the requirements of the following regulations:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, and 10 CFR Part 50 Appendix A, General Design Criteria (GDC) 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.
- GDC 2, "Design bases for protection against natural phenomena," as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions.
- GDC 4, "Environmental and dynamic effects design bases," as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions.
- GDC 10, "Reactor design," as it relates to reactor internals being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.
- GDC 14, "Reactor coolant pressure boundary," as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
- GDC 15, "Reactor coolant system design," as it relates to the reactor coolant system (RCS) being designed with a sufficient margin to ensure that design conditions are not exceeded.

The specific review areas are contained in the Standard Review Plan (SRP) Section 3.9. This review also includes the plant specific provisions of NRC Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves," as related to plant specific programs for motor-operated valves, GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," as related to the pressure locking and thermal binding for safety-related gate valves, and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," as related to the over-pressurization of isolated piping segments.

The staff reviewed the PVNGS-2 PURLR as it relates to the structural and pressure boundary integrity of the NSSS and BOP systems. Affected components in these systems include piping, in-line equipment and pipe supports, the reactor pressure vessel (RPV), reactor vessel internals (RVIs), RSGs, control element drive mechanisms (CEDMs), reactor coolant pumps (RCPs), and pressurizer.

The NSSS design at PVNGS-2 was approved by the staff via NUREG-0852, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80." The Combustion Engineering Standard Safety Analysis Report (CESSAR) describes the design of the RCS, its components, and their supports. The CESSAR describes the methodologies used to develop limiting loads and their locations, and also contains a description and analysis of the interfaces between the CE-supplied System 80 NSSS and the rest of the plant.

In evaluating the impact of the RSGs and PUR on affected components, the licensee evaluated component stresses using the original code of record in all components except the SGs. The code of record for the original steam generators (OSGs) is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Sections II, III, V, and XI, 1971 Edition, with addenda through the Winter 1973 Addenda. The RSGs were designed and fabricated to the requirements of the 1989 Edition (no addenda) of the ASME Code, Sections II, III, V, and XI.

Load evaluation consisted of establishing revised loads for the RSGs and PUR conditions, and comparing them with the Code allowables for each load case and component. The licensee applied leak-before-break (LBB) methodology to reduce the range of dynamic effects needing Code analysis for pipe breaks, and used a computer code different from that used in the original stress analyses.

With respect to LBB, the licensee applied the methodology to justify removing from consideration the dynamic effects of breaks of RCS main coolant loop (MCL) piping. The staff has previously approved the application of LBB for CESSAR NSSSs and PVNGS-2. The licensee justified the continued applicability of the LBB analyses to PVNGS-2 for the RSGs and PUR conditions. The staff's evaluation of the LBB methodology is discussed below in Section 3.4.1 of this SE.

With respect to the change in computer codes used in the stress analyses, the licensee stated that the revised analyses were performed using the ANSYS computer code. The original analyses used a group of computer codes including MEC-21, STRUDL DYNAL, and CE-DAGS. MEC-21 was used for static analyses; STRUDL DYNAL was used for dynamic seismic analyses; STRUDL DYNAL and CE-DAGS (dynamic analysis of gapped structure) were used for pipe break analyses. ANSYS is a general purpose, finite-element program with structural and heat transfer capabilities. ANSYS performs all of the pertinent STRUDL DYNAL, MEC-21, and CE-DAGS functions. As described in the PURLR, the licensee performed a benchmark of the new ANSYS computer models against the original STRUDL, MEC-21, and CE-DAGS models. The results (i.e., modal frequencies, loads, and motions) of this benchmark demonstrated the equivalence of the ANSYS analyses to the original analyses. Results from the licensee's bench marking of the ANSYS code are consistent with previous staff bench marking of ANSYS. The staff, therefore, finds the licensee's use of ANSYS acceptable for demonstrating compliance with the ASME Code limits for stress and cumulative usage factor (CUF).

### 3.1 Reactor Pressure Vessel Components

The structural analysis of the RPV is addressed in Section 5.1 of the PURLR. The stresses and CUFs were determined by analyzing the RCS at the combined RSGs and PUR conditions. The RPV was evaluated against the provisions of the ASME Code, 1971 Edition, with addenda through the Winter 1973 Addenda, which is the Code of record. The analysis considered the new dead weight and thermal loads, as well as the revised seismic and LOCA loads. The evaluation of the RCS stresses showed that the RPV analysis of record (AOR) was bounding for stresses and CUFs.

For the RPV inlet and outlet nozzles, the normal operating loads increased, the operational basis earthquake (OBE) loads decreased, and the faulted loads did not change. The licensee determined that the design and faulted condition remained bounded by the AOR. The increase in the normal loads resulted in an increase in the CUF for both the inlet and outlet nozzles, however, the CUF remains below the Code allowable.

For the closure head flange region, the evaluation showed that all loads were less than the AOR, with the exception of the faulted condition vertical load on the closure head and flange region. The analysis with the new faulted loads demonstrated that the total load on the closure head due to operating pressure, dead weight, thermal loads, and revised faulted loads, remains less than the closure stud preload; therefore, the faulted condition is not a limiting condition for the vessel closure studs. Furthermore, the maximum stresses in the head and flange region are less than the Code allowables, therefore, the head and flange region are acceptable for operation under RSGs and PUR conditions.

For the RPV inlet and outlet nozzles, the licensee determined that the PUR and RSGs condition results in an increased dead weight plus thermal load. The licensee also determined that there was a decrease in OBE loads such that the upset condition in the AOR remains bounding, however, the fatigue analyses of the nozzles were affected. The licensee revised the fatigue analyses and determined that the CUFs remain below the ASME Code limits.

The licensee also evaluated the remaining RPV components for the RSGs and PUR conditions. For the RPV nozzle supports, the licensee determined that the increases in dead weight and thermal loads were offset by decreases in OBE loads; therefore the AOR remains bounding. Reanalysis also demonstrated that the AOR remains bounding for the CEDM nozzles and incore instrumentation (ICI) nozzles. For the RPV support columns, the licensee determined that the maximum stresses remain below the Code allowable limits.

In response to the staff's request for additional information, the licensee has provided the stresses and CUFs at the limiting locations for the above components. The staff verified that the stresses and CUFs are below the Code allowables.

For the reasons set forth above, the staff concurs with the licensee's analyses and conclusion that structural loads on the RPV and its support components are within operating limits, as defined in 10 CFR 50.55a, and 10 CFR Part 50 Appendix A, for operation under RSGs and PUR conditions.



### 3.1.1 Reactor Vessel Integrity

Part 50 of 10 CFR specifies the requirements for operation of RPVs. These requirements include criteria for heat-up and cool-down limit curves, low temperature overpressure protection, pressurized thermal shock, upper shelf energy, and reactor vessel radiation surveillance capsule monitoring and testing. Operation of the RPV is contingent upon satisfying criteria for each of these requirements contained in 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

The licensee states in its PURLR that the most critical area, in terms of RPV integrity, is the beltline region of the RPV. Therefore, the changes in neutron fluence resulting from PUR were evaluated to determine the impact on RPV integrity. The licensee's evaluation indicated that the heating rates, pressure/temperature transients, and neutron fluence estimates that were used to represent operation at 3800 MWt bound the values at the PUR level of 3990 MWt. Therefore, operation at PUR level at 3990 MWt would have no detrimental impact on RPV integrity.

By letter dated June 14, 2002, the NRC staff requested the licensee to provide additional information related to RVP integrity. Specifically, the staff requested that APS discuss the effect of the PUR on pressurized thermal shock, heat-up and cool-down pressure temperature limit curves, low temperature overpressure protection, upper shelf energy, and surveillance capsule withdrawal schedule.

The licensee responded to the staff's request by letter dated August 29, 2002, and provided the following information:

The factors influencing RPV integrity are the initial properties of the materials and the neutron fluence incident on the materials. PUR does not affect the initial material properties, but the neutron fluence can change. The effect of neutron fluence changes on vessel integrity is assessed below using 10 CFR Part 50, Appendices G and H, and 10 CFR 50.61.

- a) Pressurized Thermal Shock (PTS) - The screening criteria in 10 CFR 50.61 is 270°F for plates, forgings, and axial weld material, and 300°F for circumferential weld materials. The highest  $RT_{PTS}$  value for Unit 2 at the end of the current license was determined to be 78°F for a plate from the intermediate shell course of the RPV. The projected  $RT_{PTS}$  value at the end of the current license for the Unit 2 beltline materials are summarized in Figure 7-2 of the "Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request," dated August 29, 2002. The values represent conditions for PUR (3990 MWt) given that the projected fluence at end-of-license  $3.29 \times 10^{19} \text{ n/cm}^2$ ,  $E > 1 \text{ MeV}$ , is bounded by the AOR and the method for predicting  $RT_{PTS}$  is unchanged.
- b) Vessel Fluence Evaluation - The AOR end-of-life fluence is  $3.29 \times 10^{19} \text{ n/cm}^2$  for the vessel inside surface. The AOR is based on a core power level of 4200 MWt. In 1994, WCAP-13935 (Analysis of 137 Degree

Capsule from the Arizona Public Service Company PVNGS-2 Reactor Vessel Radiation Surveillance Program) was issued. Based on that analysis, the 32 effective full power year (EFPY) peak azimuthal fluence is  $2.047 \times 10^{19}$  n/cm<sup>2</sup> for the vessel inside surface. The WCAP-13935 analysis showed that the projected end-of-life (32 EFPY) fluence was approximately one-third lower than the value in the AOR (i.e., one-third more conservative than the assessment done for the PUR submittal). The large difference between the AOR and the WCAP-13935 analysis is based on the fact that the latter did account for actual plant operation, and much of the difference is a reflection of the low leakage fuel management program employed. The PUR submittal concerning vessel fluence was based on the AOR and showed that value to be bounding.

- c) Heat-up and Cool-down Pressure Temperature Limit Curves and Low Temperature Overpressure Protection - 10 CFR Part 50, Appendix G addresses the limits on pressure and temperature that are placed on heat-up and cool-down during normal operation. There are no changes to the values used to establish the Appendix G normal operating limits. The limits represent conditions for PUR (3990 MWt) given that the projected fluence at end-of-license,  $3.29 \times 10^{19}$  n/cm<sup>2</sup>, E>1 MeV, is bounded by the AOR such that the predicted vessel material properties used to establish the heat-up and cool-down limits are unchanged. The low temperature overpressure protection limits for PUR conditions are unchanged for those same reasons.
- d) Upper Shelf Energy (USE) - 10 CFR Part 50, Appendix G requires that the upper shelf energy throughout the life of the vessel be no less than 50 ft-lb. For Unit 2, the lowest USE value at the end of the current license was determined to be 74 ft-lb from the lower shell course of the RPV. The projected USE value at the end of the current license for the Unit 2 beltline materials is summarized in Figure 7-1 of the "Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request," dated August 29, 2002. The values represent conditions for PUR (3990 MWt) given the projected fluence at end-of-life,  $3.29 \times 10^{19}$  n/cm<sup>2</sup>, E>1 MeV.
- e) Surveillance Capsule Withdrawal Schedule - 10 CFR Part 50, Appendix H defines the reactor vessel surveillance program that is to be used by the licensee to monitor the neutron radiation induced changes in fracture toughness of the vessel during the life of the plant. It includes requirements to establish a surveillance capsule withdrawal schedule. The schedule was established based on the original calculation of fluence that was shown to bound conditions for PUR (3990 MWt). The detailed surveillance schedule is discussed in UFSAR Section 5.3.1-6.6 and Table 5.3-19. Therefore, the existing surveillance capsule withdrawal schedule remains applicable under conditions for PUR.

The staff reviewed the licensee's analysis and reasoning to confirm that the 2.94 percent PUR will not have measurable effect on the RPV fracture toughness and that the RPV will remain in compliance with the requirements stated in 10 CFR 50.61 and 10 CFR Part 50, Appendix G and

Appendix H. As part of the review, the staff also reviewed the information included in its reactor vessel integrity database (RVID) for PVNGS-2. Based upon the review of information provided by the licensee and included in RVID, for the reasons set forth above, the staff determined that PVNGS-2's RPV fracture toughness will remain in compliance with 10 CFR 50.61 and 10 CFR Part 50, Appendix G and Appendix H after the 2.94 percent PUR is implemented. The staff, therefore, concludes that the 2.94 percent PUR will not have measurable effect on the RPV fracture toughness.

### **3.2 Reactor Pressure Vessel Internals**

The RVIs are addressed in Section 5.2 of the PURLR. The RVIs were evaluated for a range of loadings, including normal operating pressure and temperature differences, flow loads, vibration loads, shock loads [including OBE and safe shutdown earthquake (SSE)], loads from anticipated transients, and LOCA loads. The evaluations were performed in accordance with ASME Code, Section III, 1974 Edition with no addenda, which is the code of record for the RVI.

The PUR conditions with respect to temperatures, pressures, and flows, are bounded by the original design conditions, as shown in Table 2.1-1 of the PURLR. Since the RCS flow remains within the original design bases, the previous analysis for flow-induced vibration (FIV) remains valid. Also, the hydraulic lift forces are such that the RVIs will remain seated and stable.

The licensee's evaluation of the RVIs demonstrated that the Service Level A, B, and D stresses meet the criteria in Section III, Division 1, of the ASME Code. The licensee's evaluation also shows the deflections are such that the control element assemblies maintain their ability to scram for the RSGs and PUR conditions in accordance with the PVNGS-2 design bases.

In its September 6, 2002, response to the staff's request for additional information (RAI), the licensee provided a summary of the calculated stresses and CUFs for the RVIs. The staff verified that the stresses and CUFs are within the Code allowables. Based on this review, the staff concurs with the licensee's conclusion that the RVIs are acceptable for operation under RSGs and PUR conditions.

### **3.3 Other Equipment on Reactor Pressure Vessel**

The heated junction thermocouples (HJTCs) are described in Section 5.3 of the PURLR. The HJTCs enter the reactor head through existing CEDM nozzles. The licensee evaluated the excitation of HJTC cables and flanges for seismic and LOCA events at the RSGs and PUR conditions. For the cables, the licensee determined that the accelerations used in the qualification tests bound the RSGs and PUR conditions. For the HJTC instrument flange assemblies, the licensee determined that the original design analyses envelop the results for operation under RSGs and PUR conditions.

The incore instrumentation (ICI) tubes are described in Section 5.3 of the PURLR. The ICI tubes are attached to the lower head of the RPV and terminate at the ICI seal table. The AOR analyzed dead weight, thermal expansion, pressure, seismic, mechanical, and LOCA loads to verify the structural integrity of the tubes. The analysis used a set of configuration spectra that were intended to bound the RPV spectra and the containment basemat spectra. The licensee identified that, for the vertical direction, the original configuration spectra did not bound the basemat spectra or the RPV spectra for the RSGs and PUR conditions. (In its September 6, 2002, response to the staff's RAI, the licensee clarified that the basemat spectra had not

changed for RSGs and PUR conditions.) The licensee reanalyzed the tubes for the new spectra and determined that the AOR remained bounding for the excitation of the ICI tubes.

As described in Section 5.3.4 of the PURLR, the licensee evaluated the permanent head lift rig (HLR) structure for seismic and LOCA loads for RSGs and PUR conditions. The licensee determined that the HLR stresses were within the allowable limits for all service level conditions.

Based on our review, for the reasons set forth above, the staff concurs with the licensee's conclusion that these components remain acceptable for operation under RSGs and PUR conditions.

### **3.3.1 Control Element Drive Mechanisms**

An assessment of the CEDMs is provided in Section 5.3 of the PURLR. Additional information was provided in the licensee's September 6, 2002 response to the staff's RAI. The CEDMs were evaluated for normal conditions, upset conditions, and faulted conditions in accordance with the ASME Code, 1974 Edition with addenda through the Winter 1975 Addenda, which is the code of record.

The licensee changed the methodology by using a three-dimensional ANSYS model of the CEDMs, rather than the SAPIV code model (for response spectrum analyses) used for the AOR, and by considering LBB. The licensee benchmarked the ANSYS model by comparing the results to test data. The model was used for dead weight, seismic, and LOCA analyses. The calculated CUFs were bounded by the AOR. The stresses increased for some items, but remain less than the Code allowables.

The licensee also assessed the absolute deflections of the CEDMs. This evaluation demonstrated that (1) the control rods would trip as designed, (2) there was no impact between adjacent CEDMs, and (3) the reed switch position indicators remained qualified (e.g., deflections were less than those used in the equipment qualification tests) for the RSGs and PUR condition.

Based on the above reasoning, the staff concurs with the licensee's conclusion that the CEDMs are acceptable for operation under RSGs and PUR conditions.

### **3.4 Reactor Coolant System Components**

The RCS piping and supports are addressed in Section 5.4 of the PURLR. The licensee assessed the RCS piping and supports in accordance with the ASME Code, 1974 Edition, with addenda through the Summer 1974 Addenda, which is the code of record. The licensee assessed the RCS tributary piping and pressurizer surge line in accordance with ASME Section III, 1974 Edition, with addenda through the Winter 1975 Addenda (Summer 1979 Addenda for Subsections NB-3650 through 3680), which is the code of record for the tributary piping.

Installation of the RSGs, which are larger than the OSGs, will necessitate modification of some RCS cold leg piping. In addition, the RSGs and PUR conditions resulted in a new set of loads (i.e., pressure, thermal expansion, deadweight, seismic, and LOCA) for the RCS piping. The licensee determined that thermal design transients are bounded by the transients specified in the AOR. The seismic and LOCA loads were calculated using the ANSYS model described

above. In response to the staff's RAI, the licensee, by letter dated September 6, 2002, provided the calculated stresses and CUFs at the limiting locations. The stresses and CUFs are below the allowable limits, and therefore, RCS components were acceptable under RSGs and PUR conditions.

### **3.4.1 Leak Before Break**

During the review of CESSAR, Combustion Engineering (CE) submitted its basis for applying LBB to its System 80 NSSSs. By letter dated June 14, 1983, CE submitted a report entitled "Basis for Design of Plant Without Pipe Whip Restraints for RCS Main Loop Piping." In response to NRC staff concerns, by letter dated December 23, 1983, CE submitted a revision to the report. On the basis of deterministic fracture mechanics analyses, CE contended that postulated double-ended guillotine breaks of the NSSS MCL piping will not occur in CESSAR facilities and, therefore, did not need to be considered as a design basis for installing protective devices such as pipe whip restraints and jet impingement shields. No other changes in the design analyses were addressed. For example, no changes were proposed to the definition of a LOCA or its relationship to the regulations addressing the design requirements for the emergency core cooling systems (10 CFR 50.46), containment (GDC 16 and 50), other engineered safety features, and the conditions for environmental qualification of equipment (10 CFR 50.49).

By letter dated July 16, 1985, the licensee requested a partial exemption to GDC 4 to permit the design of PVNGS-2 without pipe whip restraints or missile shields. The technical justification for the exemption request was provided by CE letters dated June 14, 1983, and December 23, 1983, which were submitted for CESSAR. By letter dated November 3, 1984, the licensee had clarified that the technical information in the CE letters was based on PVNGS-2, which is the prototype for CESSAR (System 80) NSSSs. By letter dated November 11, 1984, the staff accepted the application of LBB for removing pipe whip restraints and missile shields in CESSAR NSSSs. By letter dated November 29, 1985, the staff granted the partial exemption to permit this application of LBB at PVNGS-2.

In evaluating the acceptability of LBB for PVNGS-2, the staff evaluated the MCL piping for the following: the location of maximum stresses in the piping, associated with the combined loads for normal operation and SSE; potential cracking mechanisms; size of throughwall cracks that would leak a detectable amount under normal loads and pressure; stability of a "leakage-size crack" under normal plus SSE loads and the expected margin in terms of load; margin based on crack size; and the fracture toughness properties of piping and weld material.

The staff's criteria for evaluation of the above parameters are delineated in Enclosure 1 to GL 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," and are as follows:

- (1) The loading conditions should include the static forces and moments (pressure, deadweight, and thermal expansion) due to normal operation, and the forces and moments associated with SSE. These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments, and safe-ends.
- (2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion

cracking, fatigue, or water hammer is not likely, should be provided. Relevant operating history should be cited, which includes systems operation procedures; system or component modification; water chemistry parameters, limits, and controls; resistance of material to various forms of stress corrosion, and performance under cyclic loadings.

- (3) A throughwall crack should be postulated at the highest stressed locations determined from (1) above. The size of the crack should be large enough so that the leakage is assured of detection with adequate margin using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.
- (4) It should be demonstrated that the postulated leakage-size crack is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. The margin, in terms of applied loads, should be determined by a crack stability analysis; i.e., that the leakage-size crack will not experience unstable crack growth even if larger loads (larger than design loads) are applied. This analysis should demonstrate that crack growth is stable and that the final crack size is limited, such that a double-ended pipe break will not occur.
- (5) The crack size margin should be determined by comparing the leakage-size crack to critical-size crack. Under normal plus SSE loads, it should be demonstrated that there is an adequate margin between the leakage-size crack and the critical-size crack to account for the uncertainties inherent in the analyses and in the leakage detection capability. A limited-load analysis may suffice for this purpose; however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferable.
- (6) The materials data provided should include types of materials and materials specifications used for base metal, weldments, and safe-ends, the materials properties including the J-R curve used in the analyses, and long-term effects such as thermal aging and other limitations to valid data (e.g., J maximum, maximum crack growth).

The staff's November 11, 1984, letter contains the staff's evaluation of LBB for CESSAR facilities. The evaluation is also reflected in Supplement 3 to NUREG-0852, dated December 1987. The staff's acceptance was based on the following:

- (1) The loads associated with the most highly stressed locations in the main loop primary system were provided and are within Code allowables.
- (2) For CE plants, there is no history of cracking failure in reactor primary coolant system piping. CE reactor coolant system primary loops have an operating history which demonstrates their inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history includes several plants with many years of operation.
- (3) The results of the leak rate calculations performed for CESSAR used initial postulated throughwall flaws that are equivalent in size to that in Enclosure 1 to GL 84-04. CESSAR facilities are expected to have an RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," so that they can detect

leakage of one gpm in one hour. The calculated leak rate through the postulated flaw is large relative to the staff's recommended sensitivity of plant leak detection systems. The margin is at least a factor of 10 on leakage.

- (4) The expected margin in terms of load for the leakage-size crack under normal plus SSE loads is greater than a factor of three when compared to the limit load. In addition, the staff found a significant margin in terms of loads larger than normal plus SSE loads.
- (5) The margin between the leakage-size crack and the critical-size crack was calculated. Again, the results demonstrated that a crack size margin of at least a factor of three exists.

The staff's November 11, 1984, letter also states that, in order for licensees with CESSAR facilities to use the above, they should confirm that their as-built facility design substantially agrees with the design described in CE's letters dated June 14 and December 23, 1983, and that the piping loads should be no greater than those cited in those documents. The licensees should also verify that the leakage detection system meets RG 1.45.

The staff's July 16, 1985, letter contains the evaluation for PVNGS-2. The LBB analysis performed by CE for CESSAR facilities was actually performed on PVNGS-2 (the prototypical CESSAR plant) using PVNGS-2 parameters. Therefore, the CE analysis envelopes the PVNGS-2 design with respect to such parameters as loads, material properties, postulated crack leakage and size, seismicity, and leak detection system capabilities. In addition, the leak detection system for PVNGS-2 is consistent with RG 1.45 so that it can detect leakage of one gpm in one hour. Based on the above, the staff found that LBB could be applied to PVNGS-2 and granted a partial exemption by letter dated November 29, 1985.

For the PUR and RSGs conditions, the licensee proposes to use LBB to determine the faulted condition loads in the RCS. In accordance with GDC 4 that was in effect during the original plant design, which did not account for LBB methodologies, the mechanical design of the PVNGS-2 RCS included postulated breaks in all high energy piping greater than one inch in diameter. With the proposed application of LBB, the dynamic effects of MCL pipe breaks were excluded from the design basis for the RCS piping and components. Following the application of LBB at PVNGS-2, the limiting pipe breaks considered in the RCS structural integrity analyses are branch line pipe breaks in the following largest tributary pipes: pressurizer surge line, safety injection lines, shutdown cooling lines, charging line, and letdown line. The licensee also analyzed the effects of the BOP pipe breaks, such as feedwater (FW) line breaks and MSLBs. No other changes in the design analyses were requested. For example, no changes were proposed to the definition of a LOCA or its relationship to the regulations addressing the design requirements for the emergency core cooling systems (10 CFR 50.46), containment (GDC 16 and 50), other engineered safety features, and the conditions for environmental qualification of electric equipment important to safety (10 CFR 50.49). The application of LBB methodology to subcompartment loads and compliance with the appropriate GDC are discussed in Section 4.2.5 of this SE.

The staff requested additional information related to the continued applicability of LBB to PVNGS-2 for the RSGs and PUR conditions after noting that the RSGs and PUR conditions involve the replacement of RCS components (e.g., SGs and cold leg elbows) and result in changes to stresses and CUFs in RCS MCL piping. The licensee responded by letter dated

September 6, 2002. Using the methodology described above, the licensee evaluated the maximum stresses and CUFs for the RSGs and PUR conditions, as well as the material properties of the RSGs and the new cold leg elbows and field welds.

The licensee stated that, for the crack that will leak at the rate of 10 gpm at normal operating conditions (leakage crack) and crack stability, the areas of concern are the RPV inlet and outlet nozzles, since these are regions of high stress. For the RSGs and PUR conditions, the bending moments for normal operating loads at the nozzles are smaller than the original bending moments used to determine the leakage crack length (defined as the length of crack that will leak 10 gpm at normal operating conditions). Consequently, the leakage crack length for the PUR condition is longer than the leakage crack for the original condition. The leakage crack for the RPV inlet and outlet nozzles is about 11 percent and 8.5 percent of the pipe circumference, respectively. The licensee stated that a critical crack of 50 percent of the pipe circumference remains stable when subjected to both normal operating and SSE loads. Thus, the leakage crack length for the RSGs and PUR configuration is below the stability criterion. Further, the combined normal operating and SSE loads for the RSGs and PUR configuration are less than those used for the original configuration, so a leaking crack in the RPV inlet or outlet nozzle will be detectable well before the crack can grow to an unstable length.

For the material property considerations, the licensee's September 6, 2002, RAI response submittal states the following:

- (1) The stress-strain curve used in the original LBB evaluation provides a reasonable representation of the nominal stress-strain properties of the MCL piping base, RSGs, and weld materials considered.
- (2) The J-R Curve for SA-516 Grade 70 plate was a good lower bound estimate for plate material. However, some weld metals tend to have an even lower toughness property. The lower bound SA-516 weld metal curve is considered a more appropriate lower bound for the MCL piping, RSGs, and weld materials being considered in this evaluation.
- (3) The original LBB analysis resulted in an acceptable margin when measured toughness properties were degraded by a factor of four. Since the weld metal lower bound toughness properties are higher than one-fourth of the toughness properties used in the original analysis, the original analysis remains conservative and valid for the lower bound weld metal J-R curve.

APS's September 6, 2002, RAI response also states that the replacement RCS piping is at the RSGs' outlet nozzles, which are not at the critical stress locations used in the LBB analysis.

In a conference call on December 12, 2002, with APS, the staff questioned why the analysis generally assumed a reduction in metal toughness by a factor of four, but did not apply this reduction to the weld metal. In its letter dated December 23, 2002, the licensee provided the following response:

At the time the Palo Verde LBB analysis was performed, there was very limited J-R Curve fracture toughness data available for the piping and weld materials. A representative J-R Curve for the SA-516 Grade 70 piping base material was selected for



use. An arbitrary factor of four was applied to this base metal curve to provide margin for the analysis, including uncertainty in material properties. The reduction was not due to any real or postulated degradation in toughness properties for the piping materials.

In the more recent evaluations performed, some weld metal J-R Curves were shown to be lower than the SA-516 Grade 70 piping base material used in the original LBB analysis. However, the RSGs and PUR LBB assessment demonstrated that the lower bound of the weld metal data was still at least a factor of two higher than the reduced J-R Curve that is the basis for the original analysis. Effectively, the reduced J-R Curve used for the original LBB analysis and the RSGs and PUR LBB assessment is a factor of 4 less than the lower bound of the base metal test data and a factor of two less than the lower bound of the weld metal test data. Therefore, the RSGs and PUR LBB assessment demonstrated that the material property curve used in the original LBB analysis is still conservative for both the base metal and weld metal piping materials and, hence the original LBB analysis remains valid.

For the reasons set forth above, the staff concurs with the licensee's conclusion that the LBB evaluation remains valid for the PVNGS-2 MCL piping. GDC 4 states that the dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. The above evaluation demonstrates that the probability of a MCL piping rupture at PVNGS-2 is extremely low for the RSGs and PUR design basis conditions; therefore, continued application of LBB to the PVNGS-2 MCL piping is acceptable.

### **3.4.2 Reactor Coolant Pumps**

The RCPs and motors are addressed in Section 5.4.5 of the PURLR. The RCPs were designed and analyzed to meet the pump design specifications and ASME Code criteria. The RCPs are designed to the 1974 Edition of the ASME Code, with no addenda.

In its assessment, the licensee determined that design transients already considered in the AOR would bound any transients applicable to the PUR. For the structural assessment, the licensee evaluated those portions of the RCPs that have relatively low stress margins (less than 10 percent), and compared the dead weight and thermal, seismic, and faulted loads for the RSGs and PUR condition to the loads in the AOR. The only location that needed reassessment of the stresses was the lower section of the motor stand shell/lower window. For this location, the stress was shown to be below the allowable limit. The new loads for the RSGs and PUR conditions were determined to be acceptable for the RCPs, and the pump pressure boundary components were demonstrated to remain within the RCP design specification and the ASME Code. Peak accelerations of RCP motors were evaluated for the RSGs and PUR conditions; calculated values were less than the design limits by significant margins for all cases.

For the reasons set forth above, the staff concurs with the licensee's conclusion that the RCPs remain acceptable for operation at the RSGs and PUR conditions.

### **3.5 Steam Generators**

The RSGs are designed and fabricated to operate at PUR conditions. Generally, the RSGs differ from the OSGs as follows: the number of tubes is increased by 10 percent and tube

material is changed from Inconel 600 to Inconel 690; the RSGs are larger, such that primary and secondary water volumes are increased, dry weight is increased, main steam and FW nozzle elevations change, and snubber lugs now project from the shell cone; new recirculation and upper blowdown nozzles are added.

The RSGs were designed and fabricated to the requirements of the ASME Code 1989 Edition (no addenda) for structural properties, thermal-hydraulic characteristics, U-bend fatigue, tube degradation, tube plugging, and repair requirements. In its September 6, 2002, response to the staff's RAI, the licensee provided the stresses and CUFs for the RSGs. The stresses and CUFs are below the Code allowables.

With respect to the potential for FIV of the SG tubes, the licensee's September 6, 2002, response to the staff's RAI states that the potential for FIV is minimized due to the design of the RSGs. FIV analyses were performed on selected tubes based on tube span parameters, such as frequency and mode shape of vibration, and fluid flow parameters, including flow velocity and fluid density. Selected tubes were modeled with the ANSYS code to determine natural frequencies and mode shapes. The FIV analysis considers fluid elastic instability and random turbulent excitation mechanisms. The evaluation results showed a maximum stability ratio of 0.38, which is less than the design goal of 0.75, in the fluid exit bend region of the tube bundle. The licensee also stated that the turbulent displacements are within the limit.

The licensee reanalyzed the loads of the sliding base assembly (baseplate and other subcomponents) using the original design methodology. The licensee accounted for the increase in dead weight of the SG, as well as the redesign of the SG skirt. The licensee stated that stresses in the vertical support structures (pads, sliding base, bolting, key ways, etc.) remain within Code limits. For the SG upper supports (i.e., snubber arrangements, upper Z keys, and supporting structures), the loads increased on some components; however, the loads remain less than the design loads.

For the reasons set forth above, the staff concurs with the licensee's conclusion that the RSGs are acceptable for operation at PUR conditions.

### **3.5.1 Steam Generator Materials**

Steam generator tubes form a part of the reactor coolant pressure boundary, to which GDC 14, "Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50 is applicable. GDC 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. As set forth in 10 CFR 50.55a, SG tubes are required to meet various specifications of the ASME Code.

The RSGs for PVNGS-2 were manufactured by Ansaldo-Camozzi Energy Special Components of Italy and the tubes were fabricated by Sandvik of Sweden. The design was performed by ABB-Combustion Engineering (which is owned by Westinghouse) based on the Combustion Engineering System 80+ design. The RSGs were designed and analyzed for the PUR conditions in accordance with the requirements of the ASME Code Section III, 1989 edition, no addenda.

The RSGs include several improvements to mitigate potential stress corrosion cracking. For example, the tubing is made of Alloy 690 thermally treated material, which has been shown to have higher resistance to corrosion cracking than the Alloy 600 material used in the original SG tubing. The replacement SG uses eggcrate configuration design to support the vertical runs of the tubes. The eggcrate design will minimize accumulation of contaminants at the tube to tube support intersections and thus minimize corrosion at those locations. Diagonal strips provide out-of-plane support to 90 degree bends and vertical grids support the horizontal run of the tubes in the upper bend region. The tube supports are made from Type 409 stainless steel which was selected for its resistance to erosion and corrosion. To mitigate potential degradation in the tubesheet, the entire length of the tube inside the tubesheet is hydraulically expanded to minimize residual stress and crevices at the top of the tubesheet.

In a letter dated June 14, 2002, the NRC staff requested additional information regarding SG tube integrity. In a letter dated August 29, 2002, the licensee responded to the staff's request as discussed below:

With respect to the structural integrity of the replacement SG tubes, the licensee will assess the integrity of the tubing using the plant's TS requirements for inspection. The evaluation, testing, and analytical processes for condition monitoring and operational assessment will be performed as specified per plant procedures. The licensee stated that it has adopted the techniques and guidance specified in Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines, and various Electric Power Research Institute (EPRI) guidelines, including SG integrity assessment guidelines, in situ pressure test guidelines, and flaw assessment handbook.

Industry operational experience is that thermally treated tubing does not undergo noticeable degradation in service. This experience, therefore, demonstrates the corrosion performance of such tubing. The licensee stated that no new forms of degradation with respect to morphology or physical characteristics are anticipated. Should tube degradation occur, it will be identified, monitored, and assessed via a TS inspection program. The program gives the frequency and sampling requirements for eddy current inspections of SG tubing. The licensee stated that the SG inspection program satisfies the requirements of ASME Section XI, 1992 edition (including Code Cases N-356, N401-1 and N402-1) and the guidance of NRC RG 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." The staff concludes that the structural integrity of the replacement SG tubes will not be negatively impacted under the PUR conditions based on the licensee's adherence to the plant's TSs.

With regard to the leakage integrity in the RSGs, the licensee stated that it uses a conservative primary-to-secondary leakage monitoring criterion that exceeds the TS requirements and EPRI guidance. The licensee uses a conservative primary-to-secondary leakage monitoring program in its plant procedures in which plant shutdown is initiated if the leakage exceeds 50 gallons per day. The leakage limit in the plant's TSs is 150 gallons per day. In addition, the leakage assessment for accident conditions does not change as a result of the RSGs or PUR. The changes resulting from PUR with respect to leakage integrity are bounded by the design basis. The staff concludes that the leakage integrity of the RSGs will not be negatively impacted under PUR conditions because the licensee's assessment showed that the leakage integrity of the RSGs is within the design basis.

With regard to the tube plugging limit, the licensee stated that the design basis for the 40 percent tube wall thickness is calculated from the margins defined in RG 1.121, "Bases for

Plugging Degraded PWR Steam Generator Tubes," and the requirements in ASME Code, Section III, NB-3324.1. The licensee calculated a tube structural limit of 0.0121 inch, which is 32 percent of the nominal tube wall thickness of 0.042 inch. The tube nominal diameter is 0.75 inch. The structural limit also satisfies the allowable radius-to-thickness ratio in ASME Section XI IWB-3521.1. The degradation limit based on the structural consideration would be 68 percent of the tube wall thickness. Considering the growth rate of potential degradation and the inspection uncertainty per RG 1.121, the staff concludes that the tube plugging limit of 40 percent in the plant's TSs is conservative and acceptable.

With regard to the corrosion resistance of Alloy 690 thermally treated tubing under the PUR conditions, the licensee stated that there is a significant database of industry literature with regard to its corrosion resistance performance. With the exception of caustic environments containing lead (unanticipated in a SG environment), Alloy 690 thermally treated material has been shown to have superior corrosion resistance over mill-annealed Alloy 600 material. Alloy 690 material has shown improvements in corrosion resistance in both primary and secondary side environments in laboratory tests. The licensee stated that for the environments bounding the PUR conditions, improvement factors of 2 to 10 times have been verified in the laboratory. Alloy 690 has been proven in nuclear plants to have resistance to primary water stress corrosion cracking because thousands of Alloy 690 tube plugs have been installed with up to 10 years of operating experience without cracking. In addition, there has not been any stress corrosion cracking identified with Alloy 690 thermally treated tubing in the last 11 years of operation. Accordingly, the staff concludes that the use of Alloy 690 thermally treated material is acceptable under the PUR conditions on the basis of laboratory tests and operating experience.

With regard to the U-bend fatigue concern, the licensee stated that the RSGs contain design and materials that preclude U-bend fatigue problems identified in NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes." Bulletin 88-02 described a tube rupture event at North Anna Unit 1 on July 15, 1987. The cause of the tube rupture at North Anna Unit 1 was high cycle fatigue in the affected tubes. The licensee stated that the tube support material for the PVNGS-2 RSGs is stainless steel A 176 Type 409. This material is not susceptible to the denting conditions experienced by the carbon steel support plates in the North Anna SGs. As discussed above, the upper tube supports in the RSGs include a combination of eggcrates, vertical strap supports, and batwing supports. The upper tube supports are designed to prevent out-of-plane deflection that would cause fatigue. The licensee stated that the upper bundle support design has about 30 years of operating experience with no evidence of fatigue concerns. As reported in Bulletin 88-02, FIV has a significant effect on tube response in cases where the fluid elastic stability ratio equals or exceeds 1.0. For the RSGs, a maximum stability ratio of 0.7 was imposed. Since the tubes in the RSGs were analyzed for various fluid velocities to ascertain that the stability ratio of 0.7 is achieved, the staff agrees with the licensee's reasoning and conclusion that the U-bend fatigue identified in Bulletin 88-02 is not applicable to the PVNGS-2 RSGs.

The RSGs have improved tube support design to address the tube wear that occurred in the OSGs. The improvements include: (1) The horizontal grids (eggcrates) provide support to the vertical runs of the tubes, (2) The vertical grids provide vertical and horizontal support to the horizontal run of tubes in the upper bend region, (3) The diagonal strips provide out-of-plane support to 90 degree tube bends, and (4) The vertical grids are welded to the diagonal supports to provide tube stability. These improvements will reduce the stresses in the tubes, and

consequently, will reduce the potential for tube wear in the RSGs. On the basis of this improved design, the staff concludes that tube wear will not be negatively impacted by the PUR conditions.

On the basis of the information the licensee provided, as set forth above, the staff concludes that the proposed PUR is acceptable with respect to RSG tubes because the structural and leakage integrity of the RSG tubes under the PUR conditions satisfy GDC 14 and the ASME Code.

### **3.6 Pressurizer**

The pressurizer is addressed in Section 5.6 of the PURLR. The pressurizer was constructed to the ASME Code, 1971 Edition, with addenda through the Winter 1973 Addenda. The licensee determined that the current design analyses for the pressurizer are bounding for the RSGs and PUR conditions; therefore, the AOR remains bounding for the pressurizer. The staff notes that the proposed RSGs and PUR conditions do not result in a change to the pressurizer operation pressure or temperature, and the temperatures of the RCS hot and cold legs remain within the original design limits. Accordingly, the staff concurs with the licensee's conclusion that the pressurizer remains acceptable for operation at RSGs and PUR conditions.

## **4.0 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT ANALYSIS**

### **4.1 Emergency Core Cooling System Performance Analysis**

The staff reviewed the ECCS performance to confirm that the system design provides an acceptable margin of safety from conditions which would lead to fuel damage during normal reactor operation including anticipated operational occurrences, and has been accomplished using acceptable analytical methods. Acceptance criteria are based on the provisions of 10 CFR 50.46.

The licensee provided LOCA analyses results for PVNGS-2 operating at 4070 MWt (3990 MWt plus 2 percent measurement uncertainty) in its December 21, 2001, PUR submittal. The staff reviewed the analyses to assure that PVNGS-2, operating at the uprated power of 3990 MWt, would satisfy the ECCS criteria of 10 CFR 50.46(b).

The licensee performed these and previous LOCA analyses using W/CE-approved LB and SBLOCA methodologies described in topical reports CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985, and CENPD-137, Supplement 1-P-A, "Calculative Methods for the CE Small Break LOCA Evaluation Model," January 1977 (S1M), respectively.

In a letter dated October 11, 2002, responding to a staff RAI, the licensee provided results of LBLOCA and SBLOCA analyses, repeating the previous results (above), and results using CENPD-132, Supplement 4-P-A, "Calculative Methods for the C-E Nuclear Power Large Break LOCA Evaluation Model," March 2001, and CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998 (S2M), respectively.

In a letter dated December 10, 2002, the licensee provided a statement that "APS and Westinghouse Electric Company, LLC, have ongoing processes that assure that LOCA analysis input values for peak cladding temperature sensitive parameters bound the as-operated plant

values for those parameters” to show that it would properly model the PVNGS-2 plant in using the above models, such that the reported results could specifically represent the PVNGS-2 ECCS performance within the applicability range of the model.

Westinghouse ZIRLO-clad fuel was first introduced in PVNGS-2 during operating Cycle 11 (April 2002). The NRC SEs of both CENPD-404-P-A and WCAP-12610-P-A discuss LOCA analyses for mixed cores with Zircaloy-clad fuel and Vantage+ (ZIRLO-clad) fuel. These SEs both agree with the statement that: “Because of the close similarity between Vantage+ and Vantage-5 (Zircaloy-clad) fuel assemblies, a mixed core penalty need not be applied to any mixed core combination of Vantage-5 (Zircaloy-clad) and Vantage+ fuel assemblies, if both have the same design features” (WCAP-12610 SE, Appendices F and G). The NRC and industry have understood that the most significant of these “design features” include geometry (e.g., differences in spacer or mixing grids) and surface roughness. These conclusions also apply to the methodologies described in CENPD-132, Supplement 3-P-A, and CENPD-137, Supplement 1-P-A. Therefore, the licensee may use any or all of the above LOCA methodologies to perform LOCA analyses for mixed cores with Zircaloy-clad fuel and ZIRLO-clad fuel.

In its letter dated October 11, 2002, the licensee provided results for PVNGS-2 LBLOCA analyses Westinghouse performed on its behalf at the PUR using both the CENPD-132, Supplement 3-P-A LBLOCA methodology and the CENPD-132, Supplement 4-P-A LBLOCA methodology. Westinghouse explicitly analyzed ZIRLO-clad fuel only with the CENPD-132, Supplement 4-P-A LBLOCA methodology.

The calculated peak cladding temperatures (PCTs), the maximum cladding oxidation (local), and the maximum core-wide cladding oxidation for both fuels using both models are given in the following table:

Model	CENPD-132-S3	CENPD-132-S4	CENPD-132-S4
Limiting Break Size/Location	0.6 DEG/PD	0.6 DEG/PD	0.8 DEG/PD
Cladding	Zirconium	Zirconium	Vantage+
PCT	2174 °F	2110 °F	2087 °F
Max. Local Oxidation	8.37%	7.6%	12.0%
Max.Total Core-Wide Oxidation (All Fuel)	<0.86%	<0.57%	<0.73%

(DEG= double-ended guillotine; PD= pump discharge.)

In its letter dated October 11, 2002, the licensee provided results of analyses Westinghouse performed on its behalf for PVNGS-2 SBLOCA at the RSG and PUR conditions using both the CENPD-137, Supplement 1-P-A (S1M) SBLOCA methodology and CENPD-137, Supplement 2-P-A (S2M) SBLOCA methodology. While the licensee states that both Zircaloy and ZIRLO-clad fuels were explicitly treated, it provided bounding values only for the S2M analysis.

In response to the staff's request during a June 16, 2003 conference call, and by letter dated July 25, 2003, the licensee provided the results of a sensitivity study it had performed similar to the sensitivity analyses presented in CENPD-137 Supplement 2-P-A, Appendix E, page 2, Response 1a. The licensee's analyses demonstrate that the S2M SBLOCA methodology applies specifically to PVNGS-2 operating at the proposed PUR. Therefore, the staff concludes that the S2M SBLOCA methodology applies specifically to PVNGS-2 operating at the proposed PUR.

Model	S1M	S2M
Limiting Break Size	0.05ft <sup>2</sup> PD	0.05ft <sup>2</sup> PD
PCT	1907 °F	1618 °F
Max. Local Oxidation	3.57%	1.28%
Max. Total Core-Wide Oxidation (All Fuel)	<0.57%	<0.2%

In its letter dated October 11, 2002, the licensee states that calculated post-LOCA oxidation for the ZIRLO-clad fuel bounds that calculated for the resident fuel despite having a lower calculated PCT. At the staff's request, the licensee also addressed the concern that the resident fuel may have preexisting oxidation that needs to be considered in estimating the total LOCA oxidation. In a letter dated August 22, 2003, the licensee provided a response to the concern, including reference to information in the CE topical report CEN-386-P-A. The staff concludes from the results of analyses identified above, and the information contained in the report referred to in the licensee's August 22, 2003, letter, that the licensee has substantiated its conclusion, stated in a July 25, 2003, letter that the LOCA analyses for PVNGS-2 operating at PUR conditions take into consideration the total LOCA oxidation and meet the oxidation criteria of 10 CFR 50.46(b)(2). Therefore, the NRC finds that the LOCA analyses for PVNGS-2 operating at PUR conditions have considered the total LOCA oxidation, including preexisting oxidation, and meet the oxidation criteria of 10 CFR 50.46(b)(2).

The staff also notes that the pre-existing oxidation of the fuel is not expected to contribute to the LOCA maximum core-wide hydrogen generation. Therefore, the staff concludes that the core-wide hydrogen generation analyses results reported above demonstrate that PVNGS-2 operating at PUR conditions meets the core-wide hydrogen generation criterion of 10 CFR 50.46(b)(3).

As discussed above, the licensee has performed LBLOCA and SBLOCA analyses for PVNGS-2 at an uprated power of 3990 MWt using approved Westinghouse/CE methodologies. The licensee's LBLOCA and SBLOCA calculations demonstrated the following:

- The calculated LBLOCA and SBLOCA values for PCT, oxidation, and core-wide hydrogen generation are less than the limits of 2200 °F, 17 percent, and 1.0 percent specified in 10 CFR 50.46(b)(1)-(3), respectively.
- Compliance with 50.46(b)(1)-(3) and (5) assures that the core will remain amenable to cooling as required by 10 CFR 50.46(b)(4). (The staff notes that other matters that could affect coolable geometry are not involved in the requested amendment.)

In summary, the NRC staff concludes that the licensee's LOCA analyses were performed with LOCA methodologies that apply to PVNGS-2 and demonstrate that PVNGS-2 complies with the requirements of 10 CFR 50.46 (b)(1)-(4). Therefore, the NRC staff finds the licensee's LOCA analyses acceptable. Compliance with the long-term cooling requirement of 10 CFR 50.46(b)(5) is discussed in Section 4.1.1 below.

#### 4.1.1 Post Loss-of-Coolant Accident Long-Term Cooling

Regulatory requirements for long-term cooling (LTC) are provided in 10 CFR 50.46(b)(5), which states, "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." In practice, following successful calculated blowdown, refill, and reflood after initiation of a LOCA, the LTC requirement will be met if the fuel cladding remains in contact with water so that the fuel cladding temperature remains essentially at or below the saturation temperature. A potential challenge to long-term cooling is that boric acid ( $\text{H}_3\text{BO}_3$ ) could accumulate within the reactor vessel, precipitate, and block water needed to keep the fuel cladding wetted by water. The staff reviewed the licensee's approach to control  $\text{H}_3\text{BO}_3$  during LTC.

The licensee's December 21, 2001, request stated that the analysis of record for long-term cooling is based on a core power of 4070 MWt (3990 MWt plus 2 percent measurement uncertainty). This is consistent with the licensee's description of LOCA analysis assumptions contained in its January 2003, Revision 11 of the UFSAR.

In its October 11, 2002, supplemental letter, the licensee stated that the analysis used for the PUR application used the  $\text{H}_3\text{BO}_3$  precipitation evaluation model described in topical report CENPD-254, "Post-LOCA Long Term Cooling Evaluation Model," dated June 1980. The staff approved this topical report by SE dated July 30, 1979. However, the staff readdressed this model in an April 24, 2002, letter to Entergy Operations titled "Arkansas Nuclear One, Unit No. 2 - Issuance of Amendment Re: Increase in Licensed Power Level." In that letter, the staff identified concerns with respect to the meaning of conservatism in determination of vessel mixing volume, the inconsistency between the Appendix K decay heat generation requirement and that of CENPD-254, and failure to include a 4 weight percent (wt%)  $\text{H}_3\text{BO}_3$  margin to account for uncertainty. The staff concluded then that these questions would be addressed on a generic basis consistent with the evaluation of CENPD-254, and that the evaluation of the Entergy model would be assessed by alternate means. In the interim, until this generic concern associated with LTC is resolved, the staff will continue to accept the licensee's emergency operating procedures (EOPs) to initiate hot leg injection within 2-3 hours of the LBLOCA.

The staff is using the same alternative approach for evaluating the PVNGS-2  $\text{H}_3\text{BO}_3$  assessment pertaining to LTC. The alternative approach addresses the probability of conditions where significant  $\text{H}_3\text{BO}_3$  accumulation may be encountered, the current staff review of the CENPD-254 assumptions, insights from predictions of significant  $\text{H}_3\text{BO}_3$  accumulation from other plants, and the licensee's EOPs to reduce the possibility that significant  $\text{H}_3\text{BO}_3$  accumulation will be encountered. There are a number of additional assumptions (not listed above) within CENPD-254 that the staff has accepted.

The staff compared the PVNGS-2 characteristics with those of other plants. This approach is similar to the staff's approach in evaluating the Arkansas Nuclear One (ANO) PUR referenced



above. This comparison is summarized in the Table 3.1. The information shows that the PVNGS-2 PUR characteristics are consistent with those of Byron/Braidwood and ANO-2 with respect to initiation of hot leg injection.

Table 3.1 - Comparison of Characteristics				
	Characteristic	Byron/ Braidwood 5% PUR	ANO-2 7.5% PUR	Requested PVNGS-2 PUR
1	Time to reach $H_3BO_3$ saturation (hours).	8.53 (5/4/01) 6.0 (4/12/02)	~2.4 to 7.3, depending on assumptions	~3.5 (UFSAR)
2	Power (MWt)	3587	3026	4070
3	Decay heat generation rate multiplier (dimensionless)	1 (5/4/01) 1.2 (4/12/02)	1.1	1.1
4	Assumed $H_3BO_3$ saturation limit (wt%)	23.53	27.6	30
5	Core plus upper plenum volume below hot leg (ft <sup>3</sup> )	1072*	940	Multiplying power by mixing volume ratio gives approximately ANO power
6	Time to hot leg injection via emergency operating procedures (hours)**	Consistent with Item 1 prediction	2 to 4	2 to 3
*Value is from NUREG-1269, Loss of Residual Heat Removal System, Diablo Canyon Nuclear Power Plant, Unit 2, April 10, 1987, June 1987. **EOPs are consistent with the sequence of events described in UFSAR Sections 6.3.2.7 and 6.3.3.				

While the staff cannot concur with the licensee's evaluation based on CENPD-254, the staff believes that there is sufficient basis to approve the license amendment, while the questions on assumptions given above are addressed on a generic basis, for the following reasons:

- The low probability of a LBLOCA where conditions leading to significant  $H_3BO_3$  accumulation may be encountered.
- The staff recently reviewed and accepted most of CENPD-254 modeling assumptions. Only a few assumptions are subject to question, and the implications of those assumptions are understood and do not invalidate the staff's finding.
- The licensee's predictions are reasonable when compared to the predictions for other plants.
- The licensee has emergency operating procedures to initiate hot leg injection within 2 to 3 hours to reduce the possibility that significant  $H_3BO_3$  accumulation will be encountered.

Based on the above, the staff concludes that the outstanding issues of certain modeling assumptions in CENPD-254 is not a significant safety concern that would prevent the staff from approving the PUR amendment. The issue exists for the plant's current operating license and the probability for the conditions leading to significant  $\text{H}_3\text{BO}_3$  accumulation are not significantly increased by the proposed amendment. Based on the four bullets above, the staff concludes that the licensee has met the regulatory requirements for LTC under 10 CFR 50.46(b)(5) with respect to the requested increase in power, and the amendment in this regard is acceptable.

## 4.2 Containment Response Analysis

The containment building is the final barrier against the release of significant amounts of radioactive fission products. The containment response analyses are performed to demonstrate compliance with 10 CFR Part 50, Appendix A, GDC 16, "Containment Design," and GDC 50, "Containment Design Basis," to demonstrate that the design pressure and temperature conditions for the containment structure are not exceeded during design-basis accidents (DBAs). In addition, a long-term pressure response analysis is performed to demonstrate compliance with GDC 38, "Containment Heat Removal."

The containment structure must be designed to withstand the pressure and temperature conditions resulting from a postulated LOCA and maintain a leaktight barrier. It must also be designed to withstand an MSLB. The analyses performed for containment response also define environmental envelopes for equipment qualification (EQ) and for mechanical and electrical equipment located within the containment. The proposed PUR and larger RSGs will both impact the containment response during DBAs.

### 4.2.1 Containment Structure

Section 6.2 of the PURLR discusses the potential impact on the containment building by RSGs and PUR conditions. The containment is designed to withstand a pressure of 60 psig and a maximum liner temperature of 300 °F. The RSGs and PUR result in an increase in the containment pressure and temperature during postulated accidents due to the following:

- (1) The power increase results in an increase in the RCS average temperature ( $T_{ave}$ ) and decay heat, which results in more energy being transferred to the containment via the LOCA break flow.
- (2) The additional RCS inventory due to the larger SGs increases the mass transferred to the containment during the LOCA blowdown.
- (3) The additional SG secondary inventory, larger heat transfer area, and a higher secondary operating pressure result in more energy being transferred to the containment for an MSLB.
- (4) The increased power results in more FW, at a higher enthalpy, being delivered to the SG secondary for release into containment during an MSLB.

The licensee's evaluation determined that the peak containment pressure and liner temperature for a LOCA are 58 psig and 270.97 °F, respectively. The peak containment pressure and liner

plate temperature for an MSLB are 41.29 psig and 252.16 °F, respectively (for containment design assessment). These remain below the design values for these containment parameters set forth in the AOR.

The staff requested additional information related to the temperatures and pressures in containment subcompartments, and whether changes in subcompartment analyses had an impact on the design basis. In its September 6, 2002, response to the staff's RAI, the licensee stated that principal subcompartments of concern are the reactor cavity, SG subcompartments, and the pressurizer subcompartment. For the pressurizer subcompartment, for which the analyses assume a double-ended break of the pressurizer surge line, the licensee determined that the energy release rates for the RSGs and PUR condition are bounded by the original plant design. For the reactor cavity and SG subcompartments, the application of LBB eliminates the dynamic effects of RCS MCL pipe breaks; therefore, the original design bounds the predicted subcompartment accident pressure. For containment temperature effects, the licensee performed a qualitative assessment that compared the RSG/PUR containment and basemat concrete thermal profiles that were generated using the COPATTA computer code to those values assumed in the original Bechtel Structural Analysis Program (BSAP) for containment internal structures. The results of this comparison determined that the original BSAP analysis remains bounding for RSGs and PUR conditions.

Based on the above reasoning, the NRC staff concurs with the licensee's conclusion that the containment building remains acceptable for operation at RSGs and PUR conditions.

#### **4.2.2 Loss-of-Coolant Accident Containment Analysis**

The changes for the proposed PUR and RSGs that have the most impact on the containment response during these DBAs are:

1. The power increase would result in an increase in the RCS average temperature and the decay heat, which results in more energy being transferred to the containment.
2. The additional RCS inventory due to the larger RSGs increases the mass being transferred to the containment.
3. The additional RSGs mass inventory, larger heat transfer area, and a higher secondary operating pressure result in more energy being transferred to the containment.

Consistent with the current design basis, three break types were investigated:

1. RCS RCP double-ended discharge leg slot break (DEDLSB)
2. RCS RCP double-ended suction leg slot break (DESLB), and
3. Double-ended hot leg slot break (DEHLSB).

All three-break locations were analyzed assuming both minimum and maximum safety injection (SI) pump flows. The limiting single failure for these analyses is a loss of one train of the CSS.

The analyses of the LOCA events were initiated from 102 percent of PUR power level, 4070 MWt. The LOCA mass and energy releases (M&E) and containment response analyses were performed in accordance with the SRP Sections 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," and 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," respectively.

The M&E for the blowdown phase of the LOCA were obtained with the previously accepted CEFLASH-4A code (Reference 1). The M&E for the reflood and post-reflood phases of the LOCA were obtained with the previously accepted methodology (Reference 2) for the cold leg breaks. The reflood and post-reflood phases were not simulated for hot leg breaks. For hot leg breaks, most of the reflood fluid does not pass through a SG before being released to the containment and there are no physical mechanisms to rapidly remove the residual SG secondary energy during or after the reflood period.

Accounting for residual stored heat including decay heat in the primary and secondary systems, the M&E for the long-term phase of the LOCA were obtained with the previously accepted CONTRANS2 containment code (Reference 3).

The containment pressure and temperature profiles were calculated using the previously accepted COPATTA code (Reference 4) based on the M&E obtained for each break location. In reviewing the licensee's initial conditions for these analyses, it was noted that the PUR analyses were based on an initial containment relative humidity of 0 percent, while the AOR, UFSAR Table 6.2.1-6, were based on 50 percent.

By letter dated August 27, 2002, in response to the staff's RAI, the licensee provided justification for the PUR analysis. The limiting relative humidity was reduced to 0 percent to provide a conservative peak pressure calculation. Consistent with SRP guidelines, the most limiting initial condition was selected for the PUR submittal. A higher initial relative humidity within containment results in a lower pressure profile and a lower peak pressure value. The additional mass of water vapor, at 50 percent relative humidity, in the containment atmosphere acts as a heat sink. A comparison of COPATTA computer runs performed by the licensee showed a reduction of 0.37 psi in the peak pressure if a value of 50 percent relative humidity was used. Therefore, assuming 0 percent relative humidity maximizes the containment pressure response and is conservative.

The NRC staff also noted that in UFSAR Table 6.2.1-6, the RCS expansion multiplier for the LOCA was stated to be 3 percent, while in the PUR submittal, the volumetric expansion multiplier value was stated to be 2 percent. In its August 27, 2002, response to the staff's RAI, the licensee indicated that UFSAR Table 6.2.1-6 is incorrect and the error would be corrected during the next update. The RCS expansion multiplier for the existing AOR, and for the PUR, is 2 percent.

The DEDLSB LOCA with maximum ECCS was identified as the pipe break with the highest peak pressure for the PUR and with RSGs. The resulting peak pressure is less than 58 psig (73 psia), and remains below the design pressure value of 60 psig. The resulting peak vapor temperature was calculated to be 308.4 °F.

The impact of PUR and RSG conditions on the peak containment liner temperature was evaluated by the licensee. The peak liner temperature was predicted to be less than 271 °F. This temperature is below the containment building liner design peak temperature of 300 °F.

Long-term analyses of the worst case DEDLSB, and of the worst-case pump suction leg break were performed by the licensee to verify the ability of the containment heat removal system (CHRS) to maintain the containment pressure and temperature below the design conditions. These evaluations were based on a conservative performance analysis of the engineered safety features. The CHRS long-term operating mode included one CSS train. The analyses showed that within 24 hours the containment pressure was reduced to less than one half of the peak containment pressure.

Based on the reasoning set forth above, the NRC staff finds there is reasonable assurance that PVNGS-2 will continue to be in compliance with GDC 16, "Containment Design," GDC 50, "Containment Design Basis," and GDC 38, "Containment Heat Removal," at the proposed PUR power level with the RSGs following a DBA LOCA.

#### **4.2.3 Main Steam Line Break Containment Analysis**

The changes for the proposed PUR and RSGs that have the most impact on the containment response during MSLB DBAs are:

1. The power increase would result in an increase in the RCS average temperature and the decay heat, which results in more energy being transferred to the SGs, and increases the severity of the MSLB blowdown to the containment.
2. The additional RSGs secondary inventory and a higher secondary operating pressure result in increased mass and additional energy being transferred to the containment.
3. The increased power results in more FW, at a higher enthalpy, being delivered to the SGs for eventual release to the containment.

Consistent with the current design basis, three power levels were investigated to evaluate the overall containment response, the containment pressure and the EQ temperature profile. These analyses were performed by the licensee at 102 percent, 75 percent, and 0 percent of the proposed PUR power level, and included the RSGs.

The MSLB M&E and containment response analyses were performed in accordance with Sections 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," and 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," respectively, of the SRP. NRC Bulletins and Information Notices were also considered as part of the licensee's analyses: Bulletin 80-04 (Reference 5) for the treatment of main FW addition, and Information Notice 84-90 (Reference 6) for the effects of main stream line breaks on EQ. The guidance provided in NUREG-0588 (Reference 7) for the treatment of EQ cases was also included in the licensee's analyses.

The previously accepted SGNIII code (Reference 8 and Reference 9) was used to generate the M&E release data for the blowdown phase of the MSLB. The analyses conservatively assumed the availability of non-emergency power. This allows for continued RCP operation and maximizes the rate of heat transfer to the affected SG maximizing the rate of mass and energy release. With non-emergency power available, an emergency diesel generator failure is not limiting and need not be postulated.

There is a main steam isolation valve (MSIV) in each of the four main steam lines. Each valve has dual solenoid valves to assure closure even with a single failure in the control system.

Single failure of the actuation signal will not prevent valve closure since both trains of main steam isolation signal actuation are provided to each MSIV. Any failure would result in the valve going to the closed position. The other MSIV isolates the unaffected SG. For the licensing analyses, the licensee assumed a random failure of an MSIV in the broken steam line. This maximizes the forward and reverse flow to the break and maximizes the consequences of the event.

There are two FW isolation valves (FWIVs) in Series in each of the two main FW lines. If one FWIV fails, the second FWIV will provide isolation. All cases were analyzed considering the flashing of fluid in the lines from the FWIVs to the affected SG and there was no need to do a separate analysis assuming FWIV failure.

The containment pressure and temperature responses to an MSLB were obtained with the previously accepted COPATTA code. The primary differences between the LOCA analysis and the MSLB analysis are:

1. For the MSLB, the M&E release calculations terminate when the affected SG dries out, and
2. The Uchida correlation is used for the heat transfer coefficient to the structural heat sinks in the MSLB, while the Tagami correlation is used for the LOCA, consistent with the guidance provided in the SRP.

In addition to these parameters, the time for CSS actuation was re-analyzed by the licensee. The existing analysis, the AOR, conservatively bounded the time to reach the containment high-high-pressure setpoint of 10 psig. For the 102 percent power case with the PUR configuration, the time to reach the pressure setpoint of 10 psig was calculated with the COPATTA code to be approximately 8 seconds. With an instrument and equipment delay time of 82 seconds, the CSS would start quenching the containment environment at about 90 seconds into the accident, or about 11.5 seconds earlier than in the existing AOR. The licensee concluded that the cumulative effect on the containment response to a DBA MSLB from the increased blowdown due to the PUR was offset by the reduction in the time for CSS actuation.

The MSLB from 102 percent of the proposed PUR rated thermal power is the limiting case for both the containment pressure and for the equipment qualification temperature profile. The peak pressure in the containment building was predicted to be 41.29 psig without the EQ analysis guidance and 41.75 psig with the EQ analysis guidance. This peak pressure is below the containment building design peak pressure of 60 psig.

No long-term analysis was performed for the MSLB since after isolation and blowdown there is no further energy input to containment. The maximum peak containment vapor temperature was calculated to be 405.55 °F and occurred at 90 seconds for the limiting 102 percent power MSLB containment design case.

The impact of the PUR and RSG conditions on the peak containment liner temperature was evaluated by the licensee. The peak liner temperature was predicted to be about 252.2 °F. This temperature is below the containment building liner design peak temperature of 300 °F.

Based on the above, the NRC staff finds that PVNGS-2 will continue to be in compliance with GDC 16, "Containment Design," and GDC 50, "Containment Design Basis," at the proposed PUR power level with RSGs following a DBA MSLB.

#### **4.2.4 Main Steam Line Break Outside Containment Analysis**

The proposed PUR and the larger SG volume and larger heat transfer area have the potential to affect the outside containment response to an MSLB for the same reasons discussed in the MSLB Containment section above. An analysis was performed by the licensee to verify the EQ temperature envelope in the main steam support structure. A 1-ft<sup>2</sup> non-mechanistic steam line break was analyzed by the licensee to quantify the effect of the PUR for this limiting break. The analyses were performed in accordance with Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," of the SRP. NRC Bulletins and Information Notices were also considered as part of the licensee's analyses: Bulletin 80-04 for the treatment of main FW addition, and Information Notice 84-90 for the effects of main steam line breaks on EQ. The guidance provided in NUREG-0588 for the treatment of EQ cases was also included in the licensee's analyses. The purpose of this analysis was to demonstrate compliance with 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases."

The break was assumed to be at the first weld outside containment. This assumption minimizes the flow resistance between the break and the affected SG and increases the calculated M&E. The M&E were generated with the EQ analysis guidelines for this event. The analyses focused on M&E releases at 102 percent power and 0 percent power. These analyses included the assumption that the MSIV in the steam line with the least flow resistance fails to close following the isolation signal. This assumption maximizes the M&E release during this event. Superheating within the SG starts when the U-tubes uncover, as specified in NRC Information Notice 84-90. The turbine stop valves were assumed to close instantaneously at the time of reactor trip. This assumption is conservative for this event because the entire steam inventory at the time of reactor trip is assumed to be forced out of the break. No leakage was assumed through the MSIVs or main FWIVs. The auxiliary feedwater (AFW) logic was assumed to function properly and to isolate all AFW to the affected SG.

There are differences between the methodology used for the new PUR analysis and the AOR methodology. These methodology differences include a reduction in some conservative input values selected in the AOR and a revised reactor trip methodology. Based on detailed analyses performed by the licensee, it was concluded that reactor trip on core protection calculator variable over-power trip and low SG pressure could be credited if the effect of the moderator temperature coefficient (MTC) was considered. The licensee's PUR analysis evaluated all reasonable reactor trips and identified the most conservative trip. The NRC staff finds the licensee's evaluation acceptable, as the resulting analysis used the most conservative reactor trip to develop conservative mass and energy release rates for the MSLB outside containment evaluation.

The AOR zero load case, the limiting case, used an MTC more negative than that allowed by the TS. This conservative MTC led to unrealistically high steam pressures during this event, preventing the actuation of the low SG pressure trip, resulting in temperatures which are higher than those achievable with the MTC allowed by the TS.

The PUR analyses were based on the MTC limit allowed by the TS, and the revised reactor trip methodology. The peak temperature in the main steam support structure based on the M&E releases calculated for the PUR are bounded by the peak temperature values generated in the AOR. The M&E released during the 0 percent PUR case was found to be bounded by the 0 percent power AOR. The NRC staff finds the licensee's use of the TS MTC limit acceptable since the MTC limit allowed by the TS is the most conservative value expected during plant operation, and the resulting analysis used this value to develop conservative M&E release rates for the MSLB outside containment evaluation.

The blowdown phase of the 102 percent power MSLB was simulated with a modified version of the SGNIII code to generate the M&E releases. The revision to the SGNIII code provided a better representation of the secondary side. The revised code provides more detailed modeling for the four main steam lines (as compared to the original analysis which modeled only two main steam lines), for the closing of the MSIVs and for the steam flow through the main steam line cross header path following the closure of MSIVs.

In response to the NRC staff's RAI, the licensee addressed the change to the SGNIII code. The change to the SGNIII code was conducted under the provisions of the Westinghouse Quality Assurance Program. This change provides the analyst with the ability to use one flow resistance and flow area through the cross header prior to the turbine stop valves closing and a different set of values following the closure of the turbine stop valves. This change provides better modeling of the steam header crossover path by providing a steaming path to the break following the closure of the turbine stop valves and prior to the MSIVs closing. The model change verification and validation are documented in a calculation performed and owned by Westinghouse. The code's response was compared to existing data and found to yield the expected results. The staff finds the change to the SGNIII code to model the four steam lines acceptable since the change only involved a better representation of the physical system without changes to either the underlying numerical solution schemes or to the conservative selection of the thermal-hydraulic models and input values used for this licensing analysis.

As a result of the change to SGNIII, after 300 seconds into the event, the PUR M&E release rate begins to decrease faster than the M&E release rate in the AOR. The temperature in the AOR continues to increase while the temperature in the PUR analysis is decreasing. The peak temperature for the PUR case occurs at 290 seconds, based on about  $44 \times 10^6$  Btu of released energy. The temperature for the AOR does not peak until 400 seconds, based on about  $49 \times 10^6$  Btu of released energy.

The subcompartment pressure and temperature model used for the PUR was identical to the existing AOR. The previously accepted PCFLUD code (Reference 10) was used to evaluate the subcompartment response to the MSLB.

The peak temperature in the main steam support structure occurred for the 0 percent power AOR case. The peak temperature was calculated to be 383 °F, and for comparison the PUR temperature was calculated to be 367 °F. For the 102 percent power case, the AOR peak temperature was calculated to be 373 °F, and for comparison the PUR temperature was calculated to be 357 °F. The licensee has determined that the EQ profile remains limited by the current AOR, the 0 percent power case. The peak pressure in the main steam support structure remains bounded by the AOR, with a calculated value about 5 psi below the design value.



Based on the above, the NRC staff finds that PVNGS-2 will continue to be in compliance with 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases," at the proposed PUR power level with the RSGs following a DBA MSLB outside containment.

#### **4.2.5 Subcompartment Loads**

The evaluation of subcompartment loads as described in SRP Section 6.2.1.2, "Subcompartment Analysis," was not specifically addressed in the PURLR. The licensee's UFSAR Section 6.2.1.1.1.1, "Containment Structure Accident Conditions," states, "These analyses were performed at 102% of Licensed Power." In UFSAR Table 6.2.1-6, the reactor power is stated to be 3954 MWt. The PUR power level at 102 percent is 4070 MWt. In response to the NRC staff's request for additional information, the licensee addressed the subcompartment loads issue by letter dated August 27, 2002.

The short-term LOCA-related M&E releases are used as input to the subcompartment analyses that are performed to ensure that the walls of a subcompartment can maintain its structural integrity during the short pressure pulse (generally less than three seconds) accompanying a high energy line pipe rupture within that subcompartment. The subcompartments evaluated include the SG compartment, the reactor cavity region, and the pressurizer compartment.

PVNGS-2 is approved to use a LBB methodology, and for the SG compartment and the reactor cavity region, LBB was used to qualitatively demonstrate that any changes associated with operation at the PUR conditions would be offset by the LBB benefit of using smaller RCS nozzle breaks. The licensee stated that the current licensing bases for these compartments remain bounding.

The pressurizer subcompartment analysis assumes a double-ended guillotine break of the pressurizer surge line. The energy release rates were calculated for the proposed PUR condition and compared with the original plant design conditions. The original plant design energy release rates continue to bound the PUR energy release rates by approximately 10 percent. Since the analysis performed for the initial plant licensing found the pressurizer subcompartment adequate, and the energy release rates for the PUR condition remain bounded by the original energy release rates, the licensee stated that the pressurizer subcompartment remains structurally acceptable.

The NRC staff finds there is reasonable assurance that PVNGS-2 will continue to be in compliance with GDC 4, "Environmental and Dynamic Effects Design Bases" and GDC 50, "Containment Design Basis," at the proposed PUR conditions since the AOR energy release rates are approximately 10 percent higher than those expected at the PUR power level and the structural integrity of the limiting subcompartment was previously found acceptable by the staff.

#### **4.2.6 Minimum Containment Pressure Analysis for ECCS Performance Capability**

The evaluation of the minimum pressure for ECCS performance as described in SRP 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," was not specifically addressed in the licensee's submittal. UFSAR Section 6.2.1.5, "Minimum Containment Pressure Analysis for ECCS Performance Capability Studies," states that "a minimum containment pressure analysis was completed in the 1995-

1996 time frame, to support an 'ECCS break spectrum' analysis for a licensed, rated thermal power of 3876 MWt. This analysis was revised in 2000, when the 'ECCS limiting break reanalysis' utilized more conservative containment heat sink values." The PUR power level at 102 percent is 4070 MWt. By letter dated August 27, 2002, in response to the NRC staff's RAI, the licensee addressed the minimum pressure for ECCS performance issue.

As stated in Section 6.1.1 of the PURLR, the existing ECCS performance analysis was performed for a core power of 4070 MWt (3990 MWt plus 2 percent measurement uncertainty). A rated core power of 3990 MWt, which is a 5 percent increase in the original licensed power of 3800 MWt, was first used in the ECCS performance analysis performed for the 2 percent stretch power license amendment request (Reference 11) and is the rated core power that has been used in all subsequent ECCS performance analyses. The fact that the stretch power analysis was performed for a 5 percent increase in power was not described in the 2 percent stretch power license amendment, nor was the description of the LBLOCA ECCS performance analysis described in the PVNGS-2 UFSAR Section 6.3.3 .

Consistent with the core power used in the LBLOCA ECCS performance analysis in UFSAR Section 6.3.3, the minimum containment pressure analysis in UFSAR Section 6.2.1.5 also used a core power of 4070 MWt. This is true for the minimum containment pressure analysis for both the ECCS break spectrum analysis and the ECCS limiting break reanalysis. Like the description of the LBLOCA ECCS performance analysis in UFSAR Section 6.3.3, this fact is also not currently described in UFSAR Section 6.2.1.5.

Since the minimum containment pressure analysis described in UFSAR Section 6.2.1.5 was performed at the uprated core power of 4070 MWt, the NRC staff finds that reanalysis of the minimum pressure for ECCS performance to demonstrate continued compliance with 10 CFR 50.46 is not necessary, rather, the current analyses are bounding for the PUR conditions.

#### **4.3 Non-LOCA Transient Analysis**

The licensee performed analyses at 4070 MWt (3990 MWt plus 2 percent measurement uncertainty) accounting for installation of the RSGs. The Combustion Engineering Nuclear Transient Simulator (CENTS) code is used for the calculation of plant response to the non-LOCA transients. The NRC staff had previously approved the CENTS code for determination of operating limits in Amendment 137 of PVNGS-2's operating license. The design-basis events (DBEs) discussed in this section are classified into three categories depending upon the expected frequency of occurrence, i.e., anticipated operational occurrences (AOOs), infrequent events, and limiting faults. The following non-LOCA transient events are discussed:

<u>UFSAR Section</u>	<u>Submittal Section</u>	<u>Transient Event</u>
15.1	6.3.1	Increase in Heat Removal by the Secondary System
15.2	6.3.2	Decrease in Heat Removal by the Secondary System
15.3	6.3.3	Decrease in RCS Flowrate

15.4	6.3.4	Reactivity and Power Distribution Anomalies
15.5	6.3.5	Increase in RCS Inventory
15.6	6.3.6	Decrease in RCS Inventory
	6.3.8	Limiting Infrequent Events

Additional assumptions are as follows: The fission source used for CENTS is ANS/ANSI 5.1-1979, which includes a  $2\sigma$  uncertainty. The minimum departure from nucleate boiling ratio (DNBR) and departure from nucleate boiling (DNB) were determined using the CETOP-D Code. For transients that include loss of RCS flow, DNBR is determined using a more detailed code called TORC. For control element assembly (CEA) ejection events, cladding and fuel integrity is simulated using the STRIKIN-II Code. The loss of RCS flow and sheared RCP shaft event is simulated using the HERMITE Code, and finally, RCS coastdown following loss of power combined with a sheared RCP shaft and a seized RCP is simulated using the COAST Code. The fission source complies with the requirements of Appendix K to 10 CFR Part 50, and therefore, is acceptable. The codes used for DNBR, core physics, and thermal-hydraulics have been approved by the NRC staff for the ranges of conditions associated with the PUR, and are acceptable.

Initial core conditions and expected instrumentation and engineered safety system response will be identified in the discussion of each transient. It should be noted that PVNGS-2 is one of the CE plants for which the anticipated transients are assumed in the licensing analysis to coincide with a single failure, thus, elevating the severity of the transient. In this manner a degree of conservatism is built into the analysis. PVNGS-2 is not equipped with power-operated relief valves.

#### **4.3.1 Increase in Heat Removal by the Secondary System**

Analyses in this section include: decrease in FW temperature, increase in FW flow, increase in steam flow, inadvertent opening of a SG relief or safety valve, and steam bypass control system (SBCS) misoperation. Two of these transients, decrease in FW temperature and increase in FW flow, are classified as AOOs and are bounded by the SBCS misoperation, which is discussed in Section 4.3.1.1, "Increased Main Steam Flow."

Excessive heat removal causes a decrease in moderator temperature, which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on the following general design criteria: (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including AOOs, (2) GDC 15, which requires that the RCS and its associated auxiliaries be designed with

appropriate margin to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) will not be exceeded during normal operations, including AOOs, (3) GDC 20, which requires that the reactor protection system be designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs, and (4) GDC 26, which requires the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including during AOOs. Specific review criteria are contained in SRP Section 15.1.1-4.

#### **4.3.1.1 Increased Main Steam Flow**

Inadvertent increased opening of the turbine admission valve (TAV) or a malfunction of the SBSCS can result in up to an 88 percent increase in nominal steam flow, which bounds an inadvertent opening either of a turbine bypass valve (TBV) or of an atmospheric dump valve (ADV). The increase in main steam flow will decrease RCS temperature and pressure, decrease SG pressure, and increase core power and heat flux.

The reactor protection system and the engineered safety system will initiate a reactor trip on high reactor power or low RCS pressure or low SG pressure. The trip signal should protect against violating any of the SAFDLs i.e., [the minimum departure from nucleate boiling ratio (MDNBR)] or the local power limits.

The acceptance criteria (as defined in the SRP Section 15.1.1) are: (1) RCS pressure should be maintained below 110 percent of the vessel design pressure for an incident of moderate frequency, (2) fuel cladding integrity should be maintained by assuring that Criterion 1 of SRP 4.4 is met (i.e., DNBR is maintained above the MDNBR value), (3) an incident of moderate frequency should not generate a more serious plant condition, and (4) an incident of moderate frequency in combination with any active component (single) failure should not result in loss of function of any fuel barrier other than cladding. A limited number of cladding failures are acceptable.

The analysis was done using the CENTS code. The CETOP-D code, DNBR with the CE-1 critical heat flux (CHF) correlation, was used for the DNBR. The system was initialized at 102 percent the power operating limit. No operator action is assumed for the first 30-minutes after transient initiation.

The increased steam flow leads to a reduction in core inlet temperature and an increase in core power, resulting in reactor trip and closure of the MSIVs. The MDNBR is 1.40, which is above the SAFDL limit of 1.34. Pressures in the RCS and the SG decrease. No other failure is induced by this transient. All of the acceptance criteria are met.

#### **4.3.1.2 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOSGADV)**

The acceptance criteria and the analysis method and codes are the same as in PURLR Section 6.3.1.3. The analysis models IOSGADV with a loss of power (LOP) following turbine trip. Because of reduced RCS temperature, the resulting power increase will cause a trip via the core protection calculator (CPC). Depending on the core burnup, it is possible for a low SG level trip to be generated earlier than the core power excursion trip. The analysis assumes that

the RSGs would reach trip setpoint pressure considerably later compared to the existing setpoint. The MDNBR value is 1.37, which is above the 1.34 acceptance limit. The maximum pressure will remain below 110 percent of the design limit. All of the acceptance criteria are satisfied.

The NRC staff has reviewed the licensee's analyses of the excess heat removal events described in Sections 4.3.1.1 and 4.3.1.2 above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed PUR level and were performed using acceptable analytical models. For the reasons set forth above, the NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, and 26 for such events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the events stated above.

#### **4.3.1.3 Steam System Piping Failures Inside and Outside Containment**

The steam release from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed responses of the reactor coolant and auxiliary systems, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, (7) core power excursion due to power demand created by excessive steam flow, (8) variables influencing neutronics, and (9) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 27, which requires that the reactivity control systems be designed, with appropriate margin for stuck rods, to ensure that the capability to cool the core is maintained, (2) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, (3) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of a propagating fracture is minimized, and (4) GDC 35, which requires that the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.1.5.

##### **4.3.1.3.1 Steam System Piping Failures Inside and Outside Containment - Mode 1 Operation**

The break of a main steam pipe causes a significant increase in steam flow and energy removal, resulting in corresponding decreases in RCS inlet temperature and pressure.

The licensee investigated a spectrum of MSLBs, i.e., with and without LOP and from full and zero power. In addition, a stuck CEA and a high-pressure injection pump failure (single failure) is assumed. One acceptance criterion is that core coolability be maintained. For PVNGS-2, the

MSLB is a limiting fault event; maximum RCS pressure should be maintained below acceptable design limits, considering potential brittle as well as ductile failures. Thus, the nil ductility temperature  $RT_{NDT}$  of 10 CFR 50.61 is also an acceptance criterion. The potential core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above SAFDL limits. If the minimum DNBR falls below SAFDL values, fuel damage would need to be assessed. Some fuel damage is acceptable as long as the core remains in place and maintains a coolable geometry.

The CENTS code was used in these evaluations with three-dimensional reactivity feedback. This capability is based on the HERMITE code, which has been approved by the NRC staff (Reference 9-42 of PURLR).

Input parameters, such as the most negative moderator and fuel feedback coefficients were chosen to maximize the possibility of return to power after shutdown.

The results show that the MSLB transient at full power with LOP is the most limiting. The reactor trip signal would be generated by the CPC due to decreasing RCP speed. Other trips that could trip the reactor in this transient are low SG pressure, high SG differential pressure, low RCS flow, and high containment pressure. The rising SG level will lead to FW isolation valve and MSIV closings.

The results also show that the RCS remains below acceptable design limits because the pressurizer safety valves (PSVs) and MSSVs will actuate, and the minimum DNBR remains above the SAFDL limit. Coolability and pressure vessel integrity are assured. Vessel protection against brittle fracture is assured because the end of life reference nil ductility temperature is 78 °F which is below the 270 °F screening criterion of 10 CFR 50.61, and therefore, all of the acceptance criteria are met.

#### **4.3.1.3.2 Steam System Piping Failures Inside and Outside Containment - Mode 3 Operation**

MSLB events at hot standby are analyzed to demonstrate adequacy of the shutdown margin. MSLB transients with and without LOP are analyzed assuming one HPSI system has failed. The remaining parameters are chosen as in 4.3.1.3.1 above, i.e., to maximize the possibility for a return to power after trip initiation. The CENTS code is supplemented by the minimum DNBR code HRISE.

The power surge resulting from the cool-down will cause a trip signal from either low SG pressure or a high log power trip. These results indicate that the case with LOP is limiting and that the acceptance criteria are met for vessel pressure and minimum DNBR. Because of enhanced cooling, the RSGs will require more shutdown margin, however, the analysis indicates that there is sufficient shutdown margin and that the plant will not return to power.

The NRC staff has reviewed the licensee's analyses of steam system piping failure events as set forth in Sections 4.3.1.3.1 and 4.3.1.3.2, and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to

ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner because the reference temperature for the nil ductility transition is 78 °F (below the 10 CFR 50.61 screening criterion), and abundant core cooling will be provided for these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 31, and 35 for these events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to steam system piping failures.

#### **4.3.1.4 Pre-Trip Main Steam Line Break Power Excursion**

During a MSLB, a damaging power spike before a reactor trip is possible. For example, if the reactor is operating at power levels less than the rated power, the power level and power-rate based shutdown signals may be delayed for a short period of time. In this situation MDNBR and fuel damage are greater concerns than exceeding the vessel pressure limits. The NRC's acceptance criteria are based on (1) GDC 10 which requires that the RCS be designed with appropriate margin to ensure that the SAFDLs are not exceeded during normal operation including AOOs, (2) GDC 15, which requires that the RCS be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded during normal operations including AOOs, and (3) GDC 26, which requires the reliable control of reactivity changes to ensure that the SAFDLs are not exceeded.

The transients examined in cases 4.3.1.3.1 and 4.3.1.3.2 above were designed to maximize the return to power after a trip. This case maximizes the power excursion before reactor trip and the possibility of DNBR challenge to fuel integrity.

The licensee's analysis was based on the CENTS code, with the initial margin and DNBR calculated using CETOP-D, which is based on the CE-1 CHF correlation. At the time of DNBR a more accurate DNBR determination is made using the TORC code.

The CPC VOPT provides an early trip for small SLBs, but for larger SLBs and limiting MTC, the power overshoot may reduce the DNBR below the limit of fuel damage. Full power was selected at 95 percent because the same thermal margin to DNB and SAFDL exists as at 100 percent, but the CPC VOPT setpoint will have room to increase in response to a power surge. The limiting case is from full (95 percent) power with offsite power available. Operator action at 30 minutes after event initiation will initiate cool-down.

The results show that the peak reactor power reaches 118.6 percent of full power and the MDNBR is 1.35. Primary and secondary pressures remain within 110 percent of design values. The screening criteria are met and the power excursion before reactor trip does not cause either the SAFDL or the RCS maximum pressure criteria to be exceeded.

The NRC staff has reviewed the licensee's analyses of the pre-trip MSLB power excursion and, for the reasons set forth above, concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, and

core cooling will be provided for this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26, following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the pre-trip MSLB power excursion.

#### **4.3.2 Decrease of Heat Removal by the Secondary System**

Loss of external load, turbine trip, inadvertent closure of MSIV, and loss of condenser vacuum will result in decrease of heat removal by the secondary system. These events result in a sudden reduction in steam flow and consequently cause RCS pressurization. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that the SAFDL are not exceeded during normal operations, including anticipated operational occurrences, (2) GDC 15, which requires that the RCS and its associated auxiliaries be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded during normal operations, including anticipated operational occurrences, and (3) GDC 26, which requires the reliable control of reactivity changes to ensure that the SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.2.1-5.

These transients are characterized as AOOs. Loss of external load will generate a turbine trip, which will result in closure of the turbine stop valves. The SBCS and the Reactor Power Cutback System can accommodate the excess steam without reactor trip. However, if these systems are in the manual mode and a turbine trip takes place, the MSSVs will limit secondary overpressure. Loss of condenser vacuum will also result in a turbine trip, which in addition will close the turbine stop valves and will trip the main FW pumps on high backpressure. Primary and secondary pressures will increase very fast, resulting in a reactor trip. The PSVs and the MSSVs will lift and keep primary and secondary pressure within limits (Note: this plant does not have power-operated relief valves). Loss of condenser vacuum is the most limiting transient for decreased heat removal by the secondary system. Loss of external load, turbine trip, inadvertent closure of the steam isolation valve, and loss of condenser vacuum were optimized to result in maximum primary and secondary pressure.

The RCS response was analyzed using the CENTS code, while DNBR was calculated using the CETOP-D code with the CE-1 CHF correlation. The system was initialized at 102 percent power. Loss of condenser vacuum was simulated with turbine trip, TAV closure and main FW flow ramping to zero. TAV closure time and main FW flow ramp period were conservatively selected to bound the actual plant configuration. High pressurizer pressure trip (HPPT) will follow with PSVs and MSSVs opening. Active single failures were considered; however, there are no single failures which would degrade the performance of the PSVs and MSSVs (Spring-loaded PSVs and MSSVs are not assumed to fail). LOP does not increase the peak power, therefore, no single failure is assumed. No operator action is assumed for the first 30 minutes after transient initiation.

The results show that the minimum DNBR remains above the original value and well above the SAFDL limit. The PSVs will open at the setpoints defined in the Technical Specifications, and the maximum primary pressure will peak at 2733 psia, i.e., below the safety limit of 2750 psia.



MSSV will open at setpoints defined in the Technical Specifications. The maximum secondary pressure will be limited to 1399 psia, which is lower than 110 percent of design pressure. Thus, the acceptance criteria have been met for fuel failure, pressure vessel maximum pressure, and SG maximum pressure.

#### **4.3.2.1 Main Steam Isolation Valve Closure**

Closure of all MSIVs could take place on a spurious signal. A decrease in heat removal results in an increases in primary and secondary temperature and pressure. Reactor trip will follow and the PSVs and the MSSVs will limit maximum pressures in the primary and the secondary system. The transient evolves as, and is bounded by, a loss of condenser vacuum.

The NRC staff has reviewed the licensee's analyses of the decrease in heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDC 10, 15, and 26 with respect to these events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the events stated.

#### **4.3.2.2 Loss of Non-Emergency AC Power to the Station Auxiliaries**

LOP to the auxiliaries can result from either LOP or loss of the onsite distribution system. On loss of Alternating current (AC) power the turbine stop valves close and FW flow to both SGs drops to zero. The RCPs will coast down and the plant will trip on DNBR. Primary and secondary pressure increases are limited from the PSVs and MSSVs. The emergency diesel generators (EDGs) will be activated to provide sufficient power for the operation of the safety equipment, including auxiliary FW. Operator control of the ADVs will regulate pressure in the SGs. This event is classified as an AOO and is bounded by the loss of condenser vacuum.

#### **4.3.2.3 Loss of Normal Feedwater Flow (LOFW)**

LOFW from the loss of one or two FW pumps will result in decreasing level and increasing pressure in the SGs. Primary temperature and pressure will rise until a reactor trip occurs from either low SG level or high pressurizer pressure. The trip signal will close the turbine stop valves, the MSSVs will control pressure in the secondary, and the low SG level will activate the AFW. The cool-down is operator controlled via the steam bypass and the condenser. This event is classified as an AOO and is bounded by the loss of condenser vacuum. Therefore, it is not reanalyzed here.

#### **4.3.2.4 Feedwater System Pipe Breaks**

Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cool-down (by excessive energy discharge through the break) or a RCS heat-up (by reducing FW flow to the affected SG). In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) the methods of

thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed response of the reactor coolant and auxiliary systems, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 27 which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (2) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, (3) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of a propagating fracture is minimized, and (4) GDC 35, which requires that the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.2.8.

For PVNGS-2, breaks up to 0.2 ft<sup>2</sup> are characterized as small and excess FW capacity is sufficient to supply the break without adverse effect on the SG feed. A large break upstream of the check valve can result in a partial or total loss of FW flow. A break downstream of the check valve has the potential to allow reverse flow from the SG. Depending on the temperature of the water in the reverse flow, an RCS cool-down or heat-up could result. Such a cool-down is bounded by the MSLB, thus, it is not considered here; therefore, the FW line break is examined as a heat-up event.

The event in its most conservative version can be described as a FW line break downstream of the check valve, loss of all FW flow, and reverse flow through the break. The MSSVs are initially open, but SG dry out and high RCS pressure will result from the transient; a low SG level trip would be generated or a high containment pressure or a high pressurizer pressure (The HPPT is the most conservative). The PSVs will lower pressure in the RCS. It is assumed that offsite power is lost several seconds after the turbine trip. The onsite diesels will be activated and the low SG level will activate AFW to refill the SGs. Operator manual cool-down commences at 30 minutes from transient initiation.

Because this is a limiting low-probability event it is classified as having a Vessel Code Service C limit, i.e., vessel pressure should be maintained below 120 percent of the design limit. DNBR should be maintained above the SAFDL limit. Fuel damage should be assessed for the part of the fuel which is calculated to fall below DNBR, and the core should remain coolable.

The CENTS code was used for the analysis, and CETOP-D was used for DNBR (incorporating the CE-1 critical heat flux correlation). Two cases were analyzed for maximum RCS pressure and core integrity and long-term cooling vs AFW capacity.

The transient simulation used limiting values to conservatively maximize RCS pressure and fuel damage. The limiting break size is conservatively determined based on the approved method, i.e., the break size is that which results in a simultaneous HPPT and Low SG Level Trip. A turbine trip will immediately follow and LOP is assumed concurrent with turbine trip. The LOP has a maximum effect on degrading RCS-to-SG heat transfer, thus maximizing RCS pressure. Single failures were considered, and the most limiting would be PSV or MSSV failure to inhibit depressurization, however, there are no credible failures for these valves. No other single failures combined with the LOP would result in a more severe transient. No operator action is assumed for the first 30 minutes following transient initiation.

The sequence of events reveals that the sudden reduction of the primary-to-secondary heat transfer caused by decrease in SG inventory and LOFW is compounded by a turbine trip caused by the rapid heat-up and the coincident LOP. However, the PSVs and the MSSVs open to provide pressure relief and cool-down. Pressure is limited to below 120 percent of the design value and the minimum DNBR is about 1.50, which is well above the limit of 1.34.

The NRC staff has reviewed the licensee's analyses of FW system pipe breaks and concludes, as described above, that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and abundant core cooling will be provided for these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 28, 31, and 35 for these events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to FW system pipe breaks.

#### **4.3.3 Decrease in Reactor Coolant Flow**

##### **4.3.3.1 Total Loss of Reactor Coolant Flow**

A decrease in reactor coolant flow or a total loss of coolant flow, occurring while the plant is at power, could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if the SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences, (2) GDC 15, which requires that the RCS and its associated auxiliaries be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded during normal operations, including anticipated operational occurrences, and (3) GDC 26, which requires that there are reliable controls of reactivity change to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.3.1-2.

Total loss of coolant flow would result from LOP to all RCPs and simultaneous turbine trip and loss of the steam dump and bypass system. Loss of coolant flow results in loss of heat transfer, increase of primary temperature and pressure and low DNBR. Depressurization is accomplished through the MSSVs and the ADVs. The major concern is the MDNBR, which occurs a few seconds after transient initiation. Because of the very short time from transient initiation to MDNBR, no single failure would make the MDNBR value smaller. Therefore, no single failure is assumed. However, initial values were chosen to minimize the DNBR.

Because this is rated as an event of moderate frequency, the maximum RCS pressure should be maintained below 110 percent of its design value. DNBR should be greater than the limiting value.

The analysis is based on the same codes and the same conservative assumptions as Section 6.3.2 of the PURLR. The results indicate that the acceptance criteria are satisfied in that the MDNBR is above the SAFDL limit; thus, cladding integrity is preserved and the maximum RCS pressure remains well below the limit of 110 percent of design value, assuring vessel integrity.

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes, as set forth above, that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 for such an event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the decrease in reactor coolant flow event.

#### **4.3.3.2 Reactor Coolant Pump Shaft Break With Loss of Offsite Power**

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a RCP. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow, while the reactor is at power, results in a degradation of core heat transfer, which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial and long-term core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 27, which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (2) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, and (3) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of propagating fracture is minimized. Specific review criteria are contained in SRP Section 15.3.3-4.

This event bounds the pump rotor seizure because in the broken shaft case the pump presents higher resistance to flow and accelerates flow decay. Therefore, pump seizure will not be discussed. The sudden stopping of the RCP causes a CPC reactor trip to be followed by a turbine trip and LOP. By including a LOP and stuck open ADV, this event is classified as a limiting fault event for PVNGS-2. The concerns are peak RCS pressure and the MDNBR value. The acceptance criteria are that the maximum RCS pressure should be maintained below acceptable design limits, and the potential for core damage is evaluated on the basis that it is

acceptable if the MDNBR remains above SAFDL limits. If the MDNBR falls below SAFDL values, fuel damage must be assessed. Some fuel damage is acceptable as long as the core remains in place and maintains a coolable geometry.

The analysis used the CENTS code and the HERMITE code to generate conditions at the time of the minimum DNBR. Then the TORC code was used to compute the actual value.

The results indicate that the peak pressure stays below the 110 percent design limit, however, the MDNBR falls below the SAFDL limit for a short period of time, thus, some cladding damage is expected. If any cladding damage occurs, however, it is limited to cladding perforation and activity release. The fuel rod retains its structural integrity and fuel coolability.

The NRC staff has reviewed the licensee's analyses of the sudden decrease in core coolant flow events and concludes, as set forth above, that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and adequate core cooling will be provided for these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27 and 31 for these events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the sudden decrease in core coolant flow events.

#### **4.3.4            Reactivity and Power Distribution Anomalies**

##### **4.3.4.1        Uncontrolled Control Element Assembly Withdrawal From a Subcritical or Low Power Condition**

An uncontrolled control element assembly withdrawal (CEAW) from subcritical or low power condition may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences, (2) GDC 20, which requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (3) GDC 25, which requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.1.

The licensee reanalyzed both the uncontrolled CEAW from subcritical and low power (hot zero power) events in support of the proposed PUR. The licensee ensured that limiting initial

conditions and input parameters were used in the analyses. The initial conditions and input parameters were varied within the ranges of the steady state operational configurations to determine a set of bounding parameters to use. These parameter ranges included instrument uncertainties and were calculated in accordance with NRC-approved reload methods listed in the COLR administrative section of the PVNGS-2 TS or documented in the PVNGS-2 UFSAR. The licensee will verify that all parameters used in the analyses remain bounding for each future reload design, in accordance with the current NRC-approved PVNGS-2 reload design methodology.

The licensee performed these analyses using the CENTS code (Reference 12) and the CETOP-D code (Reference 13). The NSSS response is simulated using CENTS, while the transient DNBR values are calculated using CETOP-D. The licensee analyzed and provided results for both the existing PVNGS-2 operating conditions and the proposed PUR conditions. The acceptance criteria for these transients are that DNBR remain above the acceptance limit and that fuel temperature remains below the fuel melt temperature. For the CEAW from subcritical event, the licensee calculated a MDNBR value of 1.60. For the CEAW from low power event, the licensee calculated a MDNBR value of 1.45. These results demonstrate that the DNBR acceptance limit of 1.34 is satisfied. The fuel temperature acceptance criterion is commonly evaluated using a peak linear heat rate (PLHR) criterion. During the PUR review the staff identified that the TS PLHR safety limit was being violated for these events. However, the licensee performed an adiabatic deposited energy calculation which demonstrated that the peak fuel temperature remains well below the limiting fuel centerline temperature for melting fuel. As a result of this review, the licensee submitted a license amendment request to change the TS safety limit from a PLHR to a fuel centerline melt temperature safety limit. This issue was resolved as part of Amendment No. 145, dated December 2, 2002 (Reference 14).

The NRC staff has reviewed the licensee's analyses of the uncontrolled CEA withdrawal from a subcritical or low power startup condition and concludes, as set forth above, that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded for these events. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 for these events following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

#### **4.3.4.2 Uncontrolled Control Element Assembly Withdrawal at Power**

An uncontrolled CEAW at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the anticipated operational occurrence and the description of the event itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including

anticipated operational occurrences, (2) GDC 20, which requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (3) GDC 25, which requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.2.

The licensee reanalyzed the uncontrolled CEAW from power event in support of the proposed PUR. The licensee ensured that limiting initial conditions and input parameters were used in the analyses. The initial conditions and input parameters were varied within the ranges of the steady state operational configurations to determine a set of bounding parameters to use. These parameter ranges included instrument uncertainties and were calculated in accordance with NRC-approved reload methods listed in the COLR administrative section of the PVNGS-2 TS or documented in the UFSAR. The licensee will verify that all parameters used in the analyses remain bounding for each future reload design, in accordance with current NRC-approved PVNGS-2 reload design methodology.

The licensee performed the analysis using the CENTS code and the CPC Fortran code. The licensee analyzed and provided results for both the existing PVNGS-2 operating conditions and the proposed PUR conditions. The termination of this event relies upon a reactor trip signal from the CPC system to ensure that the SAFDLs are not exceeded. The acceptance criteria for this transient is that the minimum DNBR remain greater than or equal to the limit, and that the fuel temperature remain less than the melting temperature. The licensee's analysis demonstrated that the acceptance criteria are not violated during this transient.

The licensee's analysis assumed an initial core inlet temperature of 548 °F, which is the lower limit of the core inlet temperature range for PVNGS-2. As part of its review, the NRC staff questioned the licensee's use of an initial core inlet temperature at the lower limit of the range (rather than the upper limit), and the impact on DNBR results. The licensee's response, dated October 11, 2002, stated that this analysis credits the CPC VOPT, and also credits the initial conditions of the event as being a known amount of thermal margin away from the DNBR SAFDL condition during normal operation. This known amount of thermal margin is maintained by PVNGS-2 TS 3.2.4. The initial parameter values used in the analysis are chosen such that the initial thermal margin is minimized. Thermal margin degradation during any given transient is calculated in terms of core power and is called the required over power margin (ROPM). The licensee states that the effect of a lower initial core inlet temperature is to delay the reactor trip, allowing the core power to increase further prior to the trip. The later reactor trip occurs due to the effect of temperature shadowing on the excore neutron detectors, which provide inputs for the CPC VOPT. The licensee stated that starting this transient from a higher initial core inlet temperature would result in an earlier CPC VOPT, which in turn would result in lower thermal degradation during the transient. The licensee provided further clarification in its letter on March 11, 2003, and demonstrated through ROM analyses that the actual value of initial core inlet temperature has a rather insignificant effect on the reactor trip system as a result of the transient, and the ROM is essentially unaffected by the selection of an initial core inlet temperature of 548 °F or 560 °F. The NRC staff has determined that the licensee's explanation is reasonable, and finds the licensee's response to be acceptable.

The NRC staff has reviewed the licensee's analyses of the uncontrolled CEAW at power event and concludes, as set forth above, that the licensee's analyses have adequately accounted for

the changes in core design needed for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded for this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 for this event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the uncontrolled control rod assembly withdrawal at power event.

#### **4.3.4.3 Single Full-Length Control Element Assembly Drop**

The NRC staff's review covered the types of control rod misoperations that are assumed to occur, including those caused by a system malfunction or operator error. The review covered (1) descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations, (2) the sequence of events, (3) the analytical model used for analyses, (4) important inputs to the calculations, and (5) the results of the analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences, (2) GDC 20, which requires that the reactor protection system be designed to automatically initiate appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (3) GDC 25, which requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.3.

As described in the UFSAR, a single full-length CEA drop results from an interruption in the electrical power to the CEDM holding coil of a single full-length CEA. The limiting case is the CEA drop, which does not cause a reactor trip to occur and results in an approach to the DNBR SAFDL.

The licensee reanalyzed the control rod misoperation event in support of the proposed PUR. The licensee ensured that limiting initial conditions and input parameters were used in the analyses. The initial conditions and input parameters were varied within the ranges of the steady state operational configurations to determine a set of bounding parameters to use. These parameter ranges included instrument uncertainties and were calculated in accordance with NRC-approved reload methods listed in the COLR administrative section of the PVNGS-2 TSs or documented in the UFSAR. During the review, the NRC staff questioned the use of an initial power level assumption of 95 percent of rated thermal power (RTP) for this event. The licensee responded to this by letter dated October 11, 2002, stating that the initial thermal margin preserved by the TS is a function of core power, and is the same between 95 percent and 100 percent of RTP. Below 95 percent RTP, additional amounts of thermal margin are preserved. The licensee determined that the limiting case for both the current and proposed power levels is the 95 percent RTP case. The licensee will verify that all parameters used in the analyses remain bounding for each future reload design, in accordance with the PVNGS-2 reload design methodology.



The licensee performed the reanalyses using the CENTS code and the CETOP-D code. The licensee analyzed and provided results for both the existing PVNGS-2 operating conditions and the proposed PUR conditions. The acceptance criteria for this transient are that DNBR remain above the acceptance limit, and fuel temperature remains below the fuel-melt temperature. The licensee provided quantitative results in its October 11, 2002, response to the staff's RAI, which demonstrated that the MDNBR value remains above the acceptance limit of 1.34 throughout the event. The licensee also provided quantitative results which demonstrate that the PLHR remains below the acceptance throughout the event.

The NRC staff has reviewed the licensee's analyses of the single full-length CEA drop event and concludes, as set forth above, that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded for this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 for this event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the single full-length CEA drop event.

#### **4.3.4.4 Startup of an Inactive Reactor Coolant Pump**

A startup of an inactive RCP transient may result in either an increased core flow or the introduction of cooler or deborated water into the core. This event causes an increase in core reactivity due to decreased moderator temperature or moderator boron concentration. The NRC staff's review covered (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including AOOs, (2) GDC 15, which requires that the RCS and its associated auxiliaries be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded during normal operations, including AOOs, (3) GDC 20, which requires that the reactor protection system be designed to automatically initiate appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs, (4) GDC 26, which requires that reliable control of reactivity changes are designed to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including AOOs, and (5) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 15.4.4-5.

The licensee reevaluated the startup of an inactive RCP transient considering the PUR conditions. The licensee concluded that reanalysis is not necessary and that the event remains bounded by the UFSAR AOR. PVNGS-2 TS preclude operation with less than all RCPs in each loop operating during Mode 1 and 2 operation. Additionally, the analysis of record examines this event for operating Modes 3 through 6 only because plant operation with less than all RCPs

running is only permitted in these modes. Therefore, the PUR will not impact the analysis of record for this event.

The NRC staff has reviewed the licensee's analyses of the startup of an inactive RCP event and, based on the foregoing, concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, 26, and 28 for this event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the startup of an inactive RCP event.

#### **4.3.4.5 Inadvertent Deboration**

Unborated water can be added to the RCS via the chemical and volume control system (CVCS). This may happen inadvertently because of operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. Operator action is needed to stop this unplanned dilution before the shutdown margin is eliminated. The NRC staff's review covered (1) conditions at the time of the unplanned dilution, (2) causes, (3) initiating events, (4) the sequence of events, (5) the analytical model used for analyses, (6) the values of parameters used in the analytical model, and (7) results of the analyses. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences, (2) GDC 15, which requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences, and (3) GDC 26, which requires that the control rods be capable of reliably controlling reactivity changes to assure that specified acceptable fuel design limits are not exceeded during normal operation, including AOOs. Specific review criteria are contained in SRP Section 15.4.6.

The licensee did not reanalyze this event, but reassessed it with respect to the PUR. The PVNGS-2 current AOR demonstrates that Mode 5 (cold shutdown) with the RCS drained results in the least time available for detection and termination of an inadvertent deboration (ID) event. Because the limiting mode is Mode 5, the PUR will not affect the current AOR conclusions. The licensee's evaluation considered the larger RSGs, which increase the RCS volume. This increase in RCS volume results in an increased time for dilution (or a decreased dilution rate), thus, the analysis of record remains bounding. Additionally, in its March 11, 2003 letter, the licensee verified that at the proposed PUR level, for Mode 1 and Mode 2 operation, the operator action time available remains bounded by the current Mode 5 AOR and that the PUR has no impact on the ID event. The NRC staff agrees that this event remains bounded by the current AOR.

The NRC staff has reviewed the licensee's analyses of the ID of the reactor coolant due to a chemical and volume control system malfunction and concludes, for the reasons set forth above, that the licensee's analyses have adequately accounted for operation of the plant at the

proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 for this event following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the decrease in boron concentration in the reactor coolant due to a chemical and volume control system malfunction.

#### **4.3.4.6 Control Element Assembly Ejection**

CEA ejection accidents cause a rapid positive reactivity insertion and an adverse core power distribution, which together could lead to localized fuel rod damage. The NRC staff evaluated the consequences of a CEA ejection accident to determine the potential damage caused to the RCPB and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The NRC staff's review covered initial conditions, rod patterns and worths, scram worth as a function of time, reactivity coefficients, the analytical model used for analyses, core parameters which affect the peak reactor pressure or the probability of fuel rod failure, and the results of the transient analyses. The NRC's acceptance criteria are based on GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria contained in SRP Section 15.4.8 and used to evaluate this accident include:

1. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.
2. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Level C" as defined in the ASME Boiler and Pressure Vessel Code.

As described in the UFSAR, a CEA ejection event results from a circumferential rupture of a CEDM housing or of the CEDM nozzle. The CEA ejection may lead to a rapid positive reactivity addition resulting in a rapid power excursion and RCS pressurization.

The licensee reanalyzed the CEA ejection event in support of the proposed PUR. To ensure that the specific acceptance criteria are satisfied at the PUR level, the licensee analyzed two cases, a fuel performance case and a peak RCS pressure case. The licensee ensured that limiting initial conditions and input parameters were used in the analyses. The initial conditions and input parameters were varied within the ranges of the steady state operational configurations to determine a set of bounding parameters to use. These parameter ranges included instrument uncertainties and were calculated in accordance with NRC-approved reload methods listed in the COLR administrative section of the PVNGS-2 TS or documented in the UFSAR. Only the high pressurizer pressure trip is credited in the analysis. Although a reactor trip on CPC or plant protection system (PPS) VOPT may occur much earlier after the initiation of the event, no credit is taken for this trip. The licensee will verify that all parameters used in the analyses remain bounding for each future reload design, in accordance with the PVNGS-2 reload design methodology.

The licensee analyzed the fuel performance case using the STRIKIN-II code (Reference 15) to simulate response of the fuel during the transient and to determine the energy deposition in the fuel. The CE-1 model in STRIKIN-II was utilized to calculate the minimum DNBR during the transient. The licensee analyzed the peak RCS pressure case using the CENTS code.

The licensee analyzed and provided results for both the existing PVNGS-2 operating conditions and the proposed PUR conditions. The licensee provided quantitative results which demonstrate that all acceptance criteria are satisfied for both cases. For the fuel performance case, the radially averaged fuel specific enthalpy is less than the 280 cal/gm at the hottest axial location of the hot fuel pin and the fuel centerline enthalpy is less than 250 cal/gm. In its October 11, 2002, response to the staff's RAI, the licensee calculated the maximum radially averaged fuel enthalpy in the hot node to be 141 cal/gm. For the peak RCS pressure case the licensee calculated a peak RCS pressure of 2702 psia, which is below the Service Level C acceptance criterion limit.

The NRC staff has reviewed the licensee's analyses of the CEA ejection accident and, as set forth above, concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage to significantly impair the capability to cool the core. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDC 28 for such events, following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the CEA ejection accident.

#### **4.3.5 Increase in Reactor Coolant Inventory**

Two potential transients involve increase of reactor coolant inventory: inadvertent activation of the core cooling system and of the pressurizer level control system concurrent with LOP, and inadvertent malfunction of the CVCS concurrent with LOP. The licensee states that these transients are bounded by the AOR. The NRC staff concurs with this assessment and deems the AOR as bounding for increase of reactor coolant inventory transients.

#### **4.3.6 Decrease in Reactor System Inventory**

Inadvertent opening of a pressurizer safety valve is discussed in Section 4.1.

##### **4.3.6.1 Double-Ended Break of a Letdown Line Outside Containment (Upstream of the Letdown Control Valve)**

RCS fluid release outside containment can result from a break in a letdown line, an instrument line, or a sample line. The letdown line is the largest and a double-ended break would yield bounding results. The flow from the letdown line is small, and although this is a LOCA, it is not treated as a LOCA because none of the phenomena associated with a LOCA are present, therefore, the focus of this analysis is on radioactive releases.

The acceptance criteria are based on GDC 55, which requires isolation valves for lines connected to RCS lines that penetrate primary containment, unless it can be demonstrated that the isolation provisions for a class of lines, such as certain small instrumentation lines, are

acceptable on some other defined basis. The allowable radioactive release from such lines is limited by the provisions of 10 CFR Part 100 exposure guidelines.

No single active failure of a containment isolation valve was considered because there are two isolation valves in Series. The analysis was based on the CENTS code, supplemented by CETOP-D for the calculation of DNBR. This being a heat-up transient, initial conditions were chosen to maximize fuel degradation. Because of several alarms, operator action was credited at 10 minutes from transient initiation by isolation of the letdown line via closing isolation valves per EOPs.

The licensee states that the results indicate that the acceptance criteria for offsite doses are met. Radioactive releases and resulting doses are evaluated in Section 4.4 of this SE. Due to the fact that this scenario is not evaluated as a LOCA, only dose calculation and radioactive release will be evaluated for a double-ended break of a letdown line outside containment, and this evaluation is contained in Section 4.4.

#### **4.3.6.2 Steam Generator Tube Rupture**

A steam generator tube rupture (SGTR) event causes a direct release of radioactive material contained in the primary coolant to the environment through the ruptured SG tube and SG safety or atmospheric relief valves. Reactor protection and engineered safety features are actuated to mitigate the accident and restrict the offsite dose to within the guidelines of 10 CFR Part 100. The NRC staff's review covered (1) postulated initial core and plant conditions, (2) method of thermal and hydraulic analysis, (3) the sequence of events (with and without offsite power available), (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions consistent with the plant's emergency operating procedures (EOPs), and (7) the results of the accident analysis. A single failure of a mitigating system is assumed for this event. The NRC staff's review for SGTR discussed in this section is focused on the thermal and hydraulic analysis for the SGTR in order to (1) support the review related to 10 CFR Part 100 for radiological consequences, which is discussed elsewhere in this SE, and (2) confirm that SGs do not experience an overfill. Preventing a SG overfill is necessary in order to prevent failure of the main steamlines. Specific review criteria are contained in SRP Section 15.6.3.

An SGTR is analyzed in the UFSAR with and without concurrent LOP. The results with LOP are bounding, thus, only this case is analyzed here. LOP is assumed to occur due to grid instability about 3 seconds after reactor trip.

Upon tube rupture, RCS water mixes with shell side water and radioactivity is transferred to the condenser via the turbine. The condenser air removal pumps will transport this activity to the atmosphere. However, because there is a concurrent LOP, there will be a turbine trip, loss of normal FW flow, loss of RCS forced circulation, and loss of condenser vacuum. With the steam bypass control system unavailable, cool-down is accomplished with AFW and through the ADVs. This, however, is a direct route for radioactive emissions to the atmosphere.

To maximize emissions, a single failure is assumed with the ADV stuck fully open. Reactor initial conditions are also chosen to maximize steam release and, therefore, atmospheric emissions.

Radiation monitors, low pressurizer level, and high SG level let the operator diagnose SGTR and trip the plant manually before reaching the reactor trip point. This will keep the ADV open for a longer period of time and maximize emissions.

The analysis of an SGTR is based on the CENTS code. The analysis covers two events, SGTR and a stuck-open ADV creating an excess steam demand. This transient is a limiting event. The plant EOPs provide operator instructions for plant recovery.

The results indicate that the behavior of the PUR plant configuration with the RSGs is similar to the existing plant configuration. EOPs are designed to preclude pressurization and challenge to the MSSVs, aid diagnosis and plant stabilization, accomplish functional recovery, provide post-tube-rupture tube coverage, maintain adequate RCS inventory, and accomplish shutdown and depressurization.

From a reactor protection point of view, the results are acceptable because the plant does not over pressurize nor does it sustain any fuel damage during the transient. As mentioned above, the radiation consequences of this event are discussed in Section 4.4 of this SE.

#### **4.3.6.2.1      Steam Generator Tube Rupture With Concurrent Loss of Offsite Power (No Stuck-Open ADV)**

As in the previous case, upon tube rupture, primary water will mix on the shell side and reach the atmosphere via the turbine, the condenser, and the condenser air removal pumps. However, SGTR will be followed by reactor and turbine trip and loss of normal FW flow, forced RCS flow, and condenser vacuum. Cool-down is maintained by using the AFW and releasing steam through the ADVs.

Radiation monitors will initiate alarms to notify the operator and aid in event diagnosis. The EOPs include explicit instructions to guide the operator to a reactor cool-down. The objectives of the EOP guidance and corresponding operator actions are: use the ADVs to control pressure and avoid challenge to the MSSVs, diagnose the event and stabilize the plant, cool-down the plant using the ADVs on both SGs before SG isolation, do a manual main steam isolation, isolate the affected SG, and cooldown and maintain adequate RCS inventory.

The results indicate that the plant will not over pressurize and the MDNBR will remain well above the SAFDL limits. (The licensee states that the atmospheric radioactivity release will be within the 10 CFR Part 100 limits.)

The NRC staff has reviewed the licensee's analysis of the SGTR accident and concludes that the licensee's analysis has adequately accounted for operation of the plant at the proposed power level and was performed using acceptable analytical methods and approved computer codes. The NRC staff further concludes that the assumptions used in this analysis are conservative and that the event does not result in an overfill of the SG. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the SGTR event.

### **4.3.7    Limiting Infrequent Events**

#### **4.3.7.1      AOOs in Combination With a Single Active Failure**

The limiting infrequent event is designed to test the plant's capability to respond to extreme transient conditions. The acceptance criteria are based on (1) GDC 10, which requires that the

RCS be designed with appropriate margin to ensure that the SAFDLs are not exceeded, (2) GDC 15, which requires that the RCS be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded, (3) GDC 26, which requires that the control rods be capable of reliably controlling reactivity changes to ensure that the SAFDLs are not exceeded, (4) GDC 27, which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (5) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, and (6) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

The licensee created a composite limiting transient to bound the MDNBR of infrequent events (including AOOs in combination with a single failure). It is assumed that the unspecified event degrades the DNBR to the SAFDL level. The most limiting single failure is LOP, resulting in the coast-down of all RCPs. This is combined with the maximum linear heat rate produced by a CEAW. No single failures or operator errors can degrade DNBR more than the above circumstances; therefore, no other failures are assumed. Initial conditions conservatively assume a 116 percent power level due to a preexisting condition from the undefined AOO and a turbine trip coincident with reactor trip, although a 3 second delay exists. No operator action is assumed for 30 minutes after transient initiation.

The acceptance criteria are those for infrequent events (including AOOs with single failure), i.e., limited fuel damage and maximum RCS pressure within 110 percent of the RCS design value.

The analysis is based on the CENTS code supplemented by CETOP-D for DNBR (using the CE-1 CHF correlation), and the HERMITE code for the calculation of the initial conditions. The MDNBR is calculated using the more detailed TORC code.

Because such events are heat-up transients, it is implicitly postulated that the PSVs will keep the maximum pressure within acceptable limits. The results are comparable to those for a broken RCP shaft, i.e., limited fuel damage and no MDNBR propagation are predicted. The maximum pressure is within acceptance limits because the PSVs have sufficient capacity to relieve overpressure. These transient analyses provides confidence that the limiting infrequent events (including AOOs in combination with a single failure) are well within prescribed limits.

The NRC staff has reviewed the licensee's analyses of the limiting hypothetical AOO transient with LOP. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system will continue to ensure that the MDNBR and the peak RCS pressure will remain within the acceptance limits for this hypothetical event. In addition, core geometry and long-term cooling will remain within acceptable limits for such an event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 26, 27, and 31 for this hypothetical event. Therefore, the NRC staff finds the proposed bounding transient acceptable.

#### **4.3.7.2 Anticipated Transient Without Scram (ATWS)**

The pressure vessel has a certain depressurization capability and a maximum pressure design value. Certain vessels have been judged to have designed pressures marginally below the maximum pressure expected to develop in an ATWS. Pressure vessels in CE plants are in this category. The ATWS rule requires an independent and diverse shutdown system. With such a system, the probability of an ATWS event is thought to be acceptably low. Therefore, the NRC staff review for an ATWS event in a CE plant is to assure that a diverse and independent shutdown system is available.

According to 10 CFR 50.62, PWRs manufactured by CE must be equipped with systems diverse and independent from the reactor trip system to scram the reactor, trip the turbine, and initiate AFW under conditions of an ATWS. The licensee indicates that the existing system does not need any modification or resetting due to the PUR and installation of the RSGs. Inasmuch as the effects of the ATWS transient depend only on the vessel peak pressure design value, the vessel venting capability, and the presence of the diverse and independent shutdown system, the PUR and the RSGs, which do not affect these items, are irrelevant to the reactor's response to an ATWS.

#### **4.3.7.3 Station Blackout**

The requirements in Section 10 CFR 50.63 provide that each light-water cooled nuclear power plant must be able to withstand a loss of all AC power. Such a loss of power is a particular concern when offsite power is interrupted. In addition, the rule provides that each plant must cope with a loss of AC power for a certain time interval depending on plant specific factors. The rule also provides that the plant may have provisions for alternate AC power supply meeting specified criteria to satisfy the requirements of 10 CFR 50.63.

In response to the station blackout (SBO) rule, PVNGS-2 provided two gas turbine-generators as an alternate AC power source. Implementation adhered to the guidance in RG 1.155, "Station Blackout." The AC source would be available within 1 hour from the initiation of the transient and be able to provide adequate power for 3 hours for a total requirement of 4 hours. The PUR was examined with respect to potential increases in needed AC power. The result indicated that all of the systems needed for decay heat removal in the four-hour post-SBO period will be adequately powered.

The NRC staff has reviewed the submitted information and concluded that the PVNGS-2 is equipped with one additional source of power in compliance with the SBO rule and adheres to the guidance in RG 1.155, which is adequate to remove decay heat. In addition, the NRC staff established that neither the PUR nor the RSGs affect the existing power source with respect to the SBO rule. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the plant's compliance with the SBO rule.

#### **4.4 Radiological Accident Evaluation**

This section of the safety evaluation addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements which the NRC staff used in reviewing the requested amendment are the accident dose guidelines in 10 CFR 100.11 and 10 CFR Part 50,



Appendix A, GDC-19, "Control Room." The staff also used the review guidance of the accident-specific criteria in Section 15 and Section 6.4 of the SRP in its review. Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in the following documents in performing this review:

- Safety Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."
- Safety Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."
- RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."
- SRP Section 6.4, "Control Room Habitability Systems."
- SRP Section 15.1.4, "Inadvertent Opening of a Steam Generator Relief or Safety Valve."
- SRP Section 15.1.5, "Steam System Piping Failures Inside and Outside Containment (PWR)," Appendix A.
- SRP Section 15.2.8, "Feedwater System Pipe Breaks Inside and Outside Containment (PWR)."
- SRP Section 15.3.3, "Reactor Coolant Pump Rotor Seizure."
- SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Appendix A.
- SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment."
- SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure."
- SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," Appendix A and Appendix B.
- SRP Section 15.7.3, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures."
- SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents."

#### **4.4.1 Accident Dose Calculations**

The NRC staff reviewed the technical analyses related to the radiological consequences of DBAs that were performed by APS in support of its proposed PUR. Information regarding these analyses was provided in Sections 6.4 and 9.9 of the licensee's PURLR, and in a supplemental letter dated September 4, 2002. The NRC staff reviewed the assumptions,

inputs, and methods used by APS to assess these impacts, and performed independent calculations to confirm the conservatism of the APS analyses.

APS performed an assessment of all significant non-LOCA and LOCA events currently analyzed in the PVNGS-2 UFSAR. These events and the corresponding SRP Sections that address them are listed below:

- Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a LOP (SRP 15.1.4)
- MSLB Outside Containment with LOP (SRP 15.1.5)
- Feedwater Line Break (FWLB) Outside Containment with LOP (SRP 15.2.8)
- Single RCP Sheared Shaft with LOP (SRP 15.3.4)
- Control Element Assembly Ejection (SRP 15.4.8)
- Double-Ended Break of a Letdown Line Outside Containment (DBLLOCUS) (SRP 15.6.2.3.2)
- SGTR with LOP and Single Failure of ADV (SRP 15.6.3)
- SGTR with LOP (SRP 15.6.3)
- LBLOCA and SBLOCA (SRP 15.6.5)
- Radioactivity Release Due to Liquid Containing Tank Failure (SRP 15.7.3)
- Fuel Handling Accidents (FHA) (SRP 15.7.4)
- Limiting Infrequent Events

APS considered the impact of the proposed changes on each of the above listed DBAs. Where the impact could not be dispositioned qualitatively (e.g., demonstrating that previous analysis bounds the proposed plant configuration), APS performed reanalysis of the impacted DBAs. For these reanalyses, APS determined the 0-2 hour exclusion area boundary (EAB) whole body and thyroid doses and the 0-30 days low population zone (LPZ) whole body and thyroid doses. APS assessed the potential doses to control room personnel from the LBLOCA, CEA ejection, RCP sheared shaft, and SGTR with LOP and single failure of an ADV. APS had previously shown these to be the limiting events with regard to control room exposures. It should be noted that the flashing fractions used for the affected generator in calculating control room doses were based on predicted primary temperature profiles.

The NRC staff's independent evaluations considered the control room dose from all of the DBAs listed above. The accident-specific sections that follow briefly describe the accident, the APS evaluation of the impact of the proposed changes including reanalyses that were performed, and the NRC staff's evaluation. Table 1 attached to this SE tabulates the analysis assumptions used by the NRC staff in its confirmatory analyses.

The magnitude of the radiological consequences of a DBA is proportional to the quantity of fission products released to the environment. This release is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point. In general, the inventory of fission products in the fuel rods, the irradiation of materials outside of the reactor and the concentration of radioactive material in the RCS are directly proportional to the rated thermal power. Thus, an increase in the rated thermal power can be expected to increase the inventory of radioactive materials available for release. The previously analyzed transport mechanisms could be affected by plant modifications associated with the PUR and by installation of the RSGs, potentially resulting in a larger release rate. Table 2.1-1 of the PURLR tabulated the significant reactor core and coolant system parameters impacted by the proposed changes. The RCS flow rate, the mass of RCS liquid, steam line pressure, and FW and steam mass flow rates are parameters for which changes could impact the transport of radioactive material. APS assessed the impact of these changes.

#### **4.4.2 IOSGADV with LOP**

This DBA postulates an inadvertent opening of a SG ADV during power operations, resulting in the release of steam from the affected SG to the environment. The affected SG will rapidly depressurize and release its inventory of dissolved radionuclides through the fully open ADV to the environment. It is assumed that the operators isolate the failed ADV 30 minutes after it opens. Since a LOP is assumed to occur with the reactor trip, the main condenser is not available as a heat sink and the unaffected SG is used to cool down the plant by dumping steam to the environment. The released steam may be contaminated due to the leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The APS analysis assumes that the RCS specific activity and RCS primary-to-secondary leakage are at the maximum values allowed by TS. Although the thermodynamic transient analyses project no fuel damage for the current fuel cycle, APS assumed 5.5 percent fuel failure to bound future fuel cycles. APS made one change in the method of evaluation of this event. Previous calculations assumed a SG iodine decontamination factor (DF) of 10 for releases from the unaffected SG. Since the SG inventory, i.e., level, is maintained, a DF of 100 for the affected SG, as used for other DBAs, is appropriate. Since this is consistent with NRC staff practice, the NRC staff finds this change in method acceptable.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ doses estimated by APS for the IOSGADV with LOP event were found to be within the acceptance criteria of the regulatory requirements. APS determined that the control room doses due to this were bounded by those estimated for other DBAs. The staff's independent calculations confirmed that the control room doses from this event would be acceptable.

#### **4.4.3 MSLB Outside Containment with LOP**

This DBA postulates an unisolable failure in one of the main steam lines at a location outside of containment, resulting in the release of steam from the affected steam line. Both SGs will rapidly depressurize through the MSLB releasing dissolved radionuclides through the faulted steam line to the environment. The MSIVs close to isolate the unaffected SG from the break. The faulted SG dries out in 30 minutes. Since a LOP is assumed to occur with the reactor trip, the main condenser is not available as a heat sink and the unaffected SG is used to cool down the plant by dumping steam to the environment via the ADVs. The released steam may be contaminated due to the leakage of reactor coolant into the SGs via small tube leaks

(i.e., primary-to-secondary leakage). The APS analysis assumes that the RCS primary-to-secondary leakage and the initial SG specific activity are at the maximum values allowed by TS. APS conservatively assumes that the initial RCS specific activity is 3.6  $\mu\text{Ci/gm}$  dose equivalent I-131, which is greater than that allowed by TS. APS considered two cases of iodine spiking. In the first case, an iodine spike is initiated by the depressurization, resulting in the release of radioiodine from the fuel at a rate 500 times the normal appearance rate. For the second case, it is assumed that an iodine spike had occurred prior to the event and that the RCS iodine specific activity is at 60  $\mu\text{Ci/gm}$  dose equivalent I-131. A SG iodine DF of 100 is assumed in the unaffected SG.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ doses estimated by APS for the MSLB with LOP event were found to be within the acceptance criteria of the regulatory requirements. APS determined that the control room doses due to this were bounded by those estimated for other DBAs. The staff's independent calculations confirmed that the control room doses from this event would be acceptable.

#### **4.4.4 FWLB Outside Containment with LOP**

This DBA postulates an unisolable failure in one of the main FW lines at a location outside of containment, resulting in the release of high energy water and steam from the affected FW line. Although there are two check valves on this line within the containment, APS conservatively assumes that both SGs will rapidly depressurize through the FWLB, releasing dissolved radionuclides to the environment, until the FW and MSIVs close to isolate the unaffected SG from the break. Since a LOP is assumed to occur with the reactor trip, the main condenser is not available as a heat sink and the unaffected SG is used to cool down the plant by dumping steam to the environment via the ADVs. The released steam may be contaminated due to the leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TS. A SG iodine DF of 100 is assumed in the unaffected SG after 30 minutes.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ doses estimated by APS for the FWLB with LOP event were found to be within the acceptance criteria of the regulatory requirements. APS determined that the control room doses due to this were bounded by those estimated for other DBAs. The NRC staff's independent calculations confirmed that the control room doses from this event would be acceptable.

#### **4.4.5 Single RCP Sheared Shaft with LOP**

This DBA postulates that a RCP shaft has suddenly sheared causing a reduction in the flow through the affected RCS loop. A reactor trip occurs, shutting down the reactor. The flow imbalance creates localized temperature and pressure changes in the core leading to localized boiling and fuel damage. APS determines the limiting product of multiplying the failed fuel fraction and the radial peaking factor to meet acceptable EAB, LPZ, and control room dose limits. APS assumes that 10 percent of the core inventory of noble gases and iodine are located in the fuel rod gap. Following the reactor trip, an operator opens a single ADV and it is

assumed to stick fully open, creating a pathway for the noble gases and iodine released from the fuel to be released to the environment. The affected SG empties and all RCS leakage into that SG is released directly to the environment. At 30 minutes after the RCP shaft shears, the operators initiate a controlled cool down using the unaffected SG. Since the main condenser is assumed to be unavailable, the plant is cooled down by releases of steam from the ADVs on the unaffected SG to the environment. Also at 30 minutes, the operators begin refilling the affected SG, covering the tubes at 90 minutes. The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TS. A SG iodine DF of 100 is assumed in the unaffected SG and, after 90 minutes, also in the affected SG. The assumption of operator actions in this event represents a change in analysis method over that used in previous calculations. The NRC staff finds that APS proposed actions and timings are reasonable in that they can all be accomplished within the control room and are addressed in plant procedures.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB, LPZ, and control room doses estimated by APS for the RCP sheared shaft with LOP event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

#### **4.4.6 CEA Ejection**

This DBA postulates the mechanical failure of a CEDM pressure housing that results in the ejection of a CEA. This failure breeches the reactor pressure vessel head and results in a LOCA to the containment. As a bounding value, APS assumes that 17 percent of the fuel rods in the core fail, instantaneously releasing all of the noble gases and iodine in the fuel rod gap. APS assumes that 10 percent of the gases are located in the fuel rod gap. A radial peaking factor of 1.77 is applied.

Two release cases are postulated:

- Release of fission products to the containment atmosphere, from where it will enter the environment via (1) the containment purge pathway, (2) containment design leakage, and (3) leakage of containment sump water outside of the containment.
- Release of fission products to the RCS with subsequent release to the environment via primary-to-secondary leakage through SGs.

APS assumes that a containment purge is in progress at the start of the event. The release is based on the release of noble gases and iodine from the RCS, assuming 1 percent failed fuel. For the containment leakage case, the containment is assumed to leak at the maximum rate allowed by TS for the first 24 hours and 50 percent of this rate for 30 days. Analysis credit was not taken for any iodine removal mechanisms. For the containment sump water leakage case, the analysis assumes that 50 percent of the iodine released from the core is dissolved in the sump water and is transported outside of the containment via the ECCSs. Leakage from these systems is assumed to occur at a rate of 3000 ml/hr. Of the iodine that leaks from these systems only 10 percent is assumed to become airborne and available for release. For the second release case, the APS analysis assumes that the RCS primary-to-secondary leakage is at the maximum value allowed by TS. A SG iodine DF of 100 is assumed in the SGs.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB, LPZ, and control room doses estimated by APS for the CEA ejection event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

#### **4.4.7 DBLLOCUS**

This DBA postulates a failure of a piping system outside of containment that provides a path for RCS dissolved noble gases and iodine to be released to the environment. The current PVNGS-2 UFSAR considered the double-ended break of a letdown line outside of containment. APS states that it reviewed all analysis parameters for impact due to the PUR and RSGs and determined that the existing analysis remains bounding. APS states that the magnitude of the releases to the environment for this DBA is primarily a function of the leak rate and the assumed RCS specific activity. The NRC staff concurs that the current analyses remain bounding and that reanalysis is not necessary. The NRC staff bases this decision on the fact that the RCS specific activity assumed in the analysis is established by TSs, which are not affected by the PUR or RSGs. Since the operating RCS pressure range per TSs remain the same and unaffected by this amendment, the leak rate will not change significantly either.

#### **4.4.8 SGTR with LOP and Single Failure of ADV**

This DBA postulates a complete severance of a single tube in one of the SGs resulting in the transfer of RCS water to the ruptured SG. The primary-to-secondary flow through the ruptured tube following an SGTR results in radioactive contamination of the secondary system. The SG pressure increases rapidly resulting in a release of contaminated steam to the environment via the MSSVs. It is assumed that the operators open the ADVs on both SGs to prevent cycling of the MSSVs. At that time, the ADV on the ruptured SG goes fully open. When this occurs, operators divert FW to the affected SG so as to maintain the SG tubes covered and thereby reduce radioactivity releases. The continuing steam releases via the fully open ADV cools down the plant. It is assumed that the releases continue for eight hours, at which time the RCS pressure reaches equilibrium with the SG pressure, stopping the release.

The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TSs. APS considered two cases of iodine spiking. In the first case, an iodine spike is initiated by the event, resulting in the release of radioiodine from the fuel at 500 times the normal appearance rate. For the second case, it is assumed that an iodine spike had occurred prior to the event and that the RCS iodine specific activity is 60  $\mu\text{Ci/gm}$  dose equivalent I-131. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The APS analysis was performed using dynamic thermo-hydraulic inputs obtained from the proprietary CENTS computer code. Since the NRC staff does not have access to the CENTS code, the NRC staff's confirming calculation was based on a low resolution extrapolation of graphs of thermo-hydraulic data provided in the transient analysis section of the PURLR. The flow versus time data in Table 1 are the values extrapolated by the NRC staff from the graphs identified by APS in its letter of September 4, 2002. The EAB, LPZ, and control room doses estimated by APS for the SGTR with LOP and single failure event were found to be within the acceptance criteria of the regulatory requirements, and therefore, are acceptable.

#### **4.4.9 SGTR with LOP**

This DBA postulates a complete severance of a single tube in one of the SGs resulting in the transfer of reactor coolant water to the ruptured SG. The primary-to-secondary flow through the ruptured tube following an SGTR results in radioactive contamination of the secondary system. The SG pressure increases rapidly resulting in a release of contaminated steam to the environment via the MSSVs. It is assumed that the operators open the ADVs on both SGs to prevent cycling of the MSSVs. Since the main condenser is assumed to be unavailable, the plant is cooled down by releases of steam from the SGs to the environment. The steaming through the ADVs on both SGs continues until the ruptured SG is isolated at 90 minutes. After the ruptured SG is isolated, steaming of the unaffected SG continues. It is assumed that the releases continue for eight hours, at which time the RCS pressure reaches equilibrium with the SG pressure, stopping the primary-to-secondary leakage.

The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TSs. APS considered two cases of iodine spiking. In the first case, an iodine spike is initiated by the event, resulting in the release of radioiodine from the fuel at a rate 500 times the normal appearance rate. For the second case, it is assumed that an iodine spike had occurred prior to the event and that the RCS iodine specific activity is 60  $\mu\text{Ci/gm}$  dose equivalent I-131.

The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The APS analysis was performed using dynamic thermo-hydraulic inputs obtained from the proprietary CENTS computer code. Since the NRC staff does not have access to the CENTS code, the NRC staff's confirming calculation was based on a low resolution extrapolation of graphs of thermo-hydraulic data provided in the transient analysis section of the PURLR. The flow versus time data in Table 1 are the values extrapolated by the NRC staff from the graphs identified by APS in its letter of September 4, 2002. APS determined that the predicted EAB and LPZ doses remain bounded by those estimated for the SGTR with LOP and Single Failure of ADV, and that the control room doses due to this were bounded by those estimated for other DBAs. Accordingly, the EAB and LPZ doses estimated by APS for the SGTR with LOP event were found to be acceptable. In addition, the NRC staff's independent calculations confirmed that the control room doses were within the acceptance criteria of the regulatory requirements and therefore acceptable.

#### **4.4.10 LBLOCA and SBLOCA**

A LOCA is a failure of the RCS that results in the loss of reactor coolant and, if not mitigated, fuel damage, possibly including a core melt, leading to substantial releases of fission products. The containment building holds up the majority of the radioactivity released from the core. Evaluation of the effectiveness of plant safety features, such as ECCS, has shown that core melt is unlikely. The objective of this DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective. Fission products from the damaged fuel are released into the RCS and then into the containment. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. A LBLOCA is postulated as the failure of the largest pipe in the RCS. A SBLOCA involves a spectrum of smaller breaks. APS assessed several small break sizes and determined, that with regard to radiological consequences, the 0.02 ft<sup>2</sup> break size was more

limiting. The APS analyses of these events were performed in a similar manner. The significant differences in the analyses are listed below:

- The timing of many events and actions in the accident scenario are a function of the break size and are derived from the transient analyses discussed in the PURLR.
- For a LBLOCA, APS assumes a core melt that releases a substantial portion of the core fission product inventory consistent with the guidance in RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." APS assumes that 100 percent of the core inventory of noble gases and 25 percent of the core inventory of iodines are available for release from the containment. For the sump case, APS assumes that 50 percent of the core inventory of iodine is in the sump. The iodine specification is 91 percent elemental, 5 percent particulate, and 4 percent organic.
- For the SBLOCA, APS assumes that fuel damage occurs, releasing the fission products in the fuel rod gaps. No core melt is projected. APS assumes that 100 percent of the gap inventory of noble gases and 25 percent of the gap inventory of iodines are available for release from the containment. For the sump case, APS assumes that 50 percent of the gap inventory of iodine is in the sump. The iodine specification is 91 percent elemental, 5 percent particulate, and 4 percent organic.

For the LBLOCA and SBLOCA, APS considered several release pathways:

- Release from unisolated containment purge line
- Release from containment leakage
- Sump water leakage from ECCS systems outside of the containment
- (LBLOCA only) Containment atmosphere bypass via depressurized SGs
- (SBLOCA only) Primary-to Secondary leakage via SGs.

For the containment purge path, APS assumes that a containment purge is in progress at the start of the event. The release is based on the release of noble gases and iodines from the RCS, assuming a pre-incident iodine spike and an RCS specific activity of 60  $\mu\text{Ci/gm}$  dose equivalent I-131.

For the containment leakage case, the containment is assumed to leak at the maximum rate allowed by TS for the first 24 hours and 50 percent of this rate for 30 days. APS credited iodine removal by containment sprays and by mechanistic plateout. The NRC staff believes that this assumption is inconsistent with the assumption that 25 percent of the core inventory of iodine is available for release. The NRC staff is of the opinion that this represents double crediting of plateout, since the 25 percent iodine available for release already reflects a reduction of 50 percent for plateout, per Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." This double crediting would reduce the level of conservatism of the analysis.



In response to a NRC staff request for justification, APS identified that the assumption had been submitted to the NRC staff in an amendment request dated June 25, 1991, and had been part of analyses performed in support of two other amendment requests. The NRC approved the requested amendments. Although the NRC staff was able to demonstrate acceptable doses without assuming mechanistic plateout in its independent analyses, it did not explicitly reject the licensee's use of the same. APS stated its position that the mechanistic plateout assumption is part of its licensing basis. Although the licensee is voluntarily requesting a PUR, the application of mechanistic plateout is independent of power and is neither ameliorated nor exacerbated by the power increase. Based on the above, the NRC staff accepts the plateout assumption for the PUR request as it is bounded by previous analysis.

For the containment atmosphere bypass via the depressurized secondary, APS had previously determined that the leakage would be equivalent to a containment leak rate of 0.9 standard cubic feet per minute (scfm). The containment source term and duration of release are as modeled for the containment leakage case above.

For the containment sump water leakage case, the analysis assumes that 50 percent of the iodine released is dissolved in the sump water and is transported outside of the containment via the ECCSs. Leakage from these systems is assumed to occur at a rate of 3000 ml/hr. Of the iodine that leaks from these systems only 10 percent is assumed to become airborne and available for release.

For the primary-to-secondary release pathway, the APS analysis assumes that the RCS primary-to-secondary leakage is at the maximum value allowed by the TS. All of the radionuclides released from the fuel gap are assumed to be dissolved in the RCS. A SG iodine DF of 100 is assumed in the SGs. The release duration is assumed to be three hours.

The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ and control room doses were estimated by APS for the LBLOCA event. The EAB and LPZ doses were estimated by APS for the SBLOCA event. The control room doses for the SBLOCA were not evaluated, since it would be bounded by the LBLOCA doses. The NRC staff performed independent calculations of the EAB, LPZ, and control room doses for the release paths that contribute the majority of the dose. The NRC staff obtained results similar to those obtained by the licensee, and the staff analyses confirmed the APS conclusions.

#### **4.4.11 Waste Gas System Failure**

For this DBA event, APS evaluated the radiological consequences of an accidental release of the maximum inventory of a waste gas decay tank as addressed in the PVNGS-2 UFSAR. APS stated that it reviewed all of the analysis parameters for impact due to the PUR and RSGs and determined that the existing analysis remains bounding. Since the maximum inventory of a waste gas decay tank is independent of reactor power and other parameter changes associated with the RSGs, the NRC staff concurs that the current analyses remain bounding and that reanalysis is not necessary.

#### **4.4.12 Radioactivity Release Due to Liquid Containing Tank Failure**

For this DBA event, APS evaluated the radiological consequences of an accidental release of maximum inventory of the refueling water tank as addressed in the PVNGS-2 UFSAR. APS

stated that it reviewed all of the analysis parameters for impact due to the PUR and RSGs and determined that the existing analysis remains bounding. The maximum inventory of a refueling water tank is independent of reactor power and other parameter changes associated with the RSGs. Accordingly, the NRC staff concurs that the current analyses remain bounding and that reanalysis is not necessary.

#### **4.4.13 Fuel-Handling Accident**

For this DBA event, APS evaluated the radiological consequences of the release of fission products due to damage of a fuel assembly during movement of irradiated fuel. APS stated that the current PVNGS-2 UFSAR analysis had been performed assuming a power level of 4070 MWt to support a license amendment request for relief from certain TSs associated with the movement of irradiated fuel. Those supporting analyses had been conservatively performed assuming a power level of 4070 MWt. The NRC approved Amendment 143 by letter dated July 25, 2002. The NRC staff concurs that the current analyses remain bounding and that further reanalysis is not necessary.

#### **4.4.14 Limiting Infrequent Events**

This DBA is a composite of infrequent events including AOOs with a single failure. The initial conditions and sequence of events were chosen to bound the infrequent events and AOOs with single failure. This event is a loss of reactor coolant flow that occurs in conjunction with a high transient reactor power level. Since a LOP is assumed to occur with the reactor trip, the main condenser is not available as a heat sink and the unaffected SG is used to cool down the plant by dumping steam to the environment via the ADVs. The released steam may be contaminated due to leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The APS analysis assumes that the RCS primary-to-secondary leakage, the initial RCS specific activity, and the initial SG specific activity are at the maximum values allowed by TSs. APS assumes that the event causes 10 percent of the fuel rods to fail, releasing their gap activity to the RCS. A SG iodine DF of 100 is assumed in both SGs.

The NRC staff performed a confirmatory analysis. The assumptions used by the staff in their confirmatory analyses are presented in Table 1. The EAB and LPZ doses estimated by APS for the AOO with LOP event were found to be within the acceptance criteria of the regulatory requirements. APS determined that the control room doses due to this were bounded by those estimated for other DBAs. The staff's independent calculations confirmed that the control room doses from this event are acceptable.

### **5.0 NUCLEAR FUEL**

#### **5.1 Core Thermal and Hydraulic Design**

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC's acceptance criteria are based on GDC 10, which requires that the reactor core be designed with appropriate

margin to assure that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs.

The licensee performed core thermal-hydraulic analyses in support of the proposed PUR. The licensee performed these analyses at the proposed updated core power level of 3,990 MWt using NRC-approved methods, consistent with the licensee's current reload analysis methodologies. In its October 11, 2002 letter, the licensee verified that all thermal-hydraulic parameters remain within the code restrictions and limitations and, in accordance with the reload methodology, will continue to do so for each future operating cycle. Steady state DNB analyses were performed using the TORC code (Reference 16), CETOP-D code, and the CE-1 CHF correlation (References 17 and 18). The licensee determined that maintaining the current TS DNBR limit of 1.34 will provide assurance at the 95/95 probability/confidence level that the hot rod will not experience DNB under PUR conditions. This DNBR limit includes appropriate penalties associated with implementing the statistical combination of uncertainties (SCU) methodology (Reference 19).

As part of the review, the NRC staff questioned the fuel rod gap conductance values the licensee assumes in the UFSAR Chapter 15 transient and accident analyses. As discussed below, the licensee provided additional information in its October 11, 2002 RAI response, which adequately addressed the NRC staff's concerns. The selection of gap conductance values can impact key transient parameter results such as pressure, temperature, and DNBR. In response to the NRC staff's questions, the licensee provided results of sensitivity studies performed to evaluate the effects of gap conductance values. The sensitivity study results demonstrated that gap conductance has a more significant impact on fuel DNBR than on system pressure results. The licensee's response stated that the calculation, selection, and use of gap conductance values are consistent with the PVNGS-2 design basis methodology currently described in the UFSAR. The licensee applies NRC-approved methods and ensures that the gap conductance values used in the analyses are conservative for the safety analyses. Additionally, as discussed in Section 4.3 of this SE, the licensee reanalyzed or evaluated the UFSAR Chapter 15 transients at the PUR conditions to verify that the DNBR limit is satisfied during normal operation and AOOs.

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed PUR on the thermal and hydraulic design of the core and the RCS. For the reasons set forth above, the NRC staff concludes that the licensee has adequately accounted for the effects of the proposed PUR on the thermal and hydraulic design and demonstrated that the design has been accomplished using acceptable analytical methods, is equivalent to proven designs, provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and is not susceptible to thermal-hydraulic instability. On this basis, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDC 10, following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to thermal and hydraulic design.

## **5.2 Core Design**

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation or

anticipated operational occurrences, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burn-up, and vessel irradiation. The NRC's acceptance criteria are based on (1) GDC 10, which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences, (2) GDC 11, which requires that the core and associated coolant system be designed so as to assure that the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity, (3) GDC 12, which requires that the reactor core and associated coolant system be designed so as to preclude power oscillations which could result in conditions exceeding specified acceptable fuel design limits, (4) GDC 13, which requires that instrumentation be designed to monitor variables and systems affecting the fission process over their anticipated ranges for normal operation, for anticipated operational occurrences and for accident conditions, and that controls be provided for maintaining the variables and systems within prescribed operating ranges, (5) GDC 20, which requires that the reactor protection system be designed to initiate automatically operation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions, (6) GDC 25, which requires that the protection system be designed so that a single malfunction of the reactivity control system does not cause a violation of the specified acceptable fuel design limits, (7) GDC 26, which requires two independent reactivity control systems of different design, with one system using control rods and being capable of reliably controlling reactivity changes to assure that SAFDLs are not exceeded during normal operation including AOOs, and the second system having the capability to control the rate of reactivity changes resulting from planned, normal power changes, (8) GDC 27, which requires the reactivity control systems to be designed to have the capability in conjunction with poison addition by the ECCS, to reliably control reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, and (9) GDC 28, which requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding, and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 4.3, "Nuclear Design."

In support of the PUR, the licensee provided information which demonstrates that the typical uprate core design will be similar to existing core designs for PVNGS-2. The licensee provided a comparison of several key core design parameters for a PUR core design. The licensee utilizes NRC-approved core design methodology (Reference 16) for current core designs and will continue to apply this same methodology for future uprated power core designs. In accordance with this approved methodology, the licensee verifies that all key parameters remain within code restrictions and limitations for each reload cycle. The licensee will continue to perform cycle-specific core design analyses and to verify the applicability of the assumptions and parameters to future reload cycles, including the uprated unit reloads, in accordance with the approved reload process and methods. Additionally, as discussed in Section 4.3 of this SE, the licensee reanalyzed or evaluated the UFSAR Chapter 15 transients at the PUR conditions to verify that the acceptance criteria limits are satisfied during normal operation and AOOs.

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed PUR on the nuclear design. Because the licensee demonstrated that the typical uprate core design using currently approved methodology will be similar to existing core designs, the NRC staff concludes that the licensee has adequately accounted for the effects of the proposed PUR on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients. Similarly, the licensee has demonstrated that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs 10, 11, 12, 13, 20, 25, 26, 27, and 28. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the nuclear design.

### **5.3 Fuel System Design**

The fuel system consists of fuel rods, spacer grids, guide thimbles, top and bottom end plates, and reactivity control rods, including burnable poison rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is needed, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, failure mechanisms, and safety of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46 for core cooling, (2) GDC 10, which requires that the reactor core and associated coolant and control systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences, (3) GDC 27, which requires that the reactivity control system be designed to have the capability, in conjunction with the ECCS, of controlling reactivity changes to assure that the capability to cool the core is maintained under accident conditions with appropriate margin for stuck rods, and (4) GDC 35, which requires that an ECCS be provided to transfer heat from the reactor core following any loss of reactor coolant above certain rates. Specific review criteria are contained in SRP Section 4.2.

#### **5.3.1 Cladding Collapse**

If axial gaps in the fuel column occur, the cladding has the potential to be flattened by the large external coolant system pressure, a phenomenon called cladding collapse. A flattened cladding is considered a failed rod because of the large local strain. The fuel rod is designed to preclude cladding collapse.

The licensee analyzed the cladding collapse condition using the approved CEPAN code to calculate the collapse time as a function of residence time. The result showed that the rod collapse time is greater than the planned residence time for PUR conditions. This indicates that cladding collapse will not occur for the fuel rod design. Since the licensee used the approved methodology within its approved range of applicability with appropriate PUR typical inputs, the staff concurs with the licensee's cladding collapse analysis for PVNGS-2 under PUR conditions.

### **5.3.2 Strain Fatigue**

The fuel rod strain fatigue capability could be impacted by PUR conditions of higher operating temperatures and cyclic strains. The approved analysis of strain fatigue is based on the O'Donnell and Langer curve as described in the SRP Section 4.2.

The licensee reanalyzed the strain fatigue capability under PUR conditions using the O'Donnell and Langer curve. The result showed that the fuel system design maintained its strain fatigue capability. Since the licensee used the approved analysis within its approved range of applicability with appropriate PUR typical inputs, the NRC staff concludes that the strain fatigue capability is acceptable for PVNGS-2 under PUR conditions.

### **5.3.3 Clad Stress and Strain**

SRP Section 4.2 states that the stress and strain limits in fuel designs should not be exceeded for normal operations and AOOs. During PUR conditions, the fuel system could experience high-power duty loading for certain AOOs, thereby exceeding the stress and strain limits.

The licensee reexamined the fuel system loading using the approved methodologies for calculating stress and strain limits. The results showed that the stress and strain limits were not exceeded for PUR conditions. Since the licensee used the approved methodologies within their approved ranges of applicability and with appropriate PUR typical values of input parameters, the NRC staff concludes that the fuel system design meets the stress and strain limits for PVNGS-2 under PUR conditions.

### **5.3.4 Rod Internal Pressure**

Rod internal pressure is considered a driving force for fuel system damage that could contribute to the loss of dimensional stability and cladding integrity. The NRC staff has approved topical report for Westinghouse (formerly CE) fuel designs CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," in which a rod pressure limit can exceed the system pressure provided that the fuel to cladding gap remains closed, i.e., there is no clad-liftoff.

The rod internal pressure will increase during PUR conditions. The licensee performed a bounding analysis using the approved fuel performance code FATES3B. The result showed that the maximum predicted rod pressure was below the critical pressure limit of no clad liftoff. Since the licensee used the approved methodologies within their approved ranges of applicability and with appropriate PUR typical values of input parameters, the staff considers that the rod internal pressure analysis is acceptable for PVNGS-2 under PUR conditions. However, because the methodology has been validated only up to a peak rod average burnup of 60,000 MWD/MTU, this conclusion is subject to that burnup limit.

### **5.3.5 Cladding Oxidation**

SRP Section 4.2 identifies cladding oxidation buildup as a potential damage mechanism for fuel designs. The SRP further states that the effect of cladding oxidation needs to be addressed in safety and design analyses such as in the thermal and mechanical analyses. Recently the

NRC staff realized that, in order to maintain adequate cladding ductility at high burnups, the total amount of oxidation or corrosion should be limited during normal operations, including AOOs. The licensee has adopted a corrosion limit of 100 microns for ZIRLO-clad fuel in its UFSAR.

The cladding corrosion will conceivably increase during PUR conditions. The licensee performed a bounding corrosion analysis that showed the maximum corrosion was within the 100 microns limit under PUR conditions. Based on the acceptable corrosion result, the staff concludes that the impact of corrosion on the thermal and mechanical performance will be minimal for PVNGS-2 under PUR conditions.

### **5.3.6 Fuel System Design Conclusion**

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed PUR on the fuel system design. For the reasons set forth above, the NRC staff concludes that the licensee has adequately accounted for the effects of the proposed PUR on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. On this basis, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC 10, GDC 27, and GDC 35 following implementation of the proposed PUR. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the fuel system design for peak rod average burnup of up to 60,000 MWD/MTU.

### **5.4 Neutron Fluence**

The NRC staff reviewed the licensee's fluence calculation methodology, including input parameters and approximations used. Fluence values determine reactor pressure boundary material properties which are then used to calculate pressure-temperature limits for cold overpressure protection and pressurized thermal shock. The acceptance criteria are based on GDCs 14, 30 and 31. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," also describes acceptable methods for vessel fluence calculations; however, the AOR for PVNGS-2 dates back to its initial safety analysis, before the issuance of RG 1.190, and is the basis for the design of the reactor pressure vessel.

The licensee performed fluence calculations in the existing analysis of record at the 4200 MWt power level. In addition, the out-in type of fuel loading was assumed, however, the proposed PUR is for 3990 MWt and the loading pattern has been for low leakage since cycle 2 for PVNGS-2. Both conditions are conservative and the AOR bounds the values calculated for the PUR. PVNGS-2 has a calculated vessel fluence value which has been reviewed and approved with the approval of the 32 EFPY pressure temperature curves of amendment 117 dated May 20, 1998. The staff concludes that the fluence values of record are conservative and therefore, are acceptable.

## **6.0 BALANCE-OF-PLANT**

### **6.1 Balance of Plant Components**

The licensee described the BOP systems and components evaluation in Section 8 of the PURLR. The licensee evaluated the safety-related BOP piping, components, and supports affected by the RSGs and PUR operation in accordance with the provisions of the ASME Code, 1974 Edition, with addenda through the Winter 1975 Addenda, which is the code of record. The evaluation considered the normal, upset, emergency, and faulted conditions. The licensee provided the maximum stresses in the affected BOP piping in the September 6, 2002, response to the NRC staff's RAI. The stresses are below the Code allowables.

The BOP components (e.g., heat exchangers, pumps, and valves) that are most affected by the RSGs and PUR are those of the main steam system. The affected components include the MSSVs, MSIVs, turbine stop valves, turbine throttle valves, low pressure FW heaters, high pressure FW heaters, condensate pumps, FW pumps, and heater drain pumps. The RSGs and PUR conditions do not raise the main steam operating pressure above the original design pressure. The licensee found that the original component design analyses bound the predicted flow rates, temperatures, and pressures for the RSGs and PUR conditions.

The PUR results in an increase in main steam flow and FW flow. The NRC staff requested additional information related to the potential for FIV in the affected BOP components. In its September 6, 2002, response to the NRC staff's RAI, the licensee stated that all secondary side component steam and water velocities are predicted to remain below the original component design values when operating at PUR conditions. The licensee also stated that the post-PUR startup testing will be conducted in accordance with ASME OM-S/G-2000, "Standards and Guides for Operation and Maintenance of Nuclear Power Plants," including Part 3, "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems."

The NRC staff noted that the acceptability of several secondary system components (i.e., steam traps) relies on an improvement in the steam quality (from 0.25 percent to 0.1 percent) to offset the increase in steam flow under the PUR conditions. In its September 6, 2002, response to the NRC staff's RAI, the licensee stated that the improved steam quality is a performance criterion for the RSGs, and the steam quality will be measured during the post-PUR startup test program. The licensee also stated that the measured steam quality for the OSGs is currently higher than 0.25 percent (which is the design value for the OSGs), such that the anticipated 0.1 percent for the RSGs will result in a more than four-fold reduction in total moisture carryover. Since the secondary system capacities are not challenged at the current operating conditions, the licensee concludes that they will be adequate for the RSGs and PUR conditions.

The licensee's programs for valves are discussed in Section 9.5 of the PURLR. The licensee reviewed the design basis of the safety-related power-operated valves, including motor, air, and solenoid operated valves considering GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." The design parameters included pressure, temperature, and differential pressure. The licensee concludes that:



- (1) There are no changes to the design basis pressure or temperature of any safety-related power-operated valves.
- (2) The existing valve actuators will adequately perform their intended design function after PUR, with the exception of the main steam isolation valve bypass valves (MSIVBVs), which require a modification to operate with increased differential pressure.

In its September 6, 2002, response to the NRC staff's RAI, the licensee stated that the MSIVBVs have a design function to close on a main steam isolation signal and are normally closed during full power operation. The actuators are being modified such that the valves are capable of closing under maximum, worst-case differential pressure loads. The modifications consist of increasing the pre-load of the actuator stanchion springs, increasing the pressure of the air that opens the valves (by removing the regulator), and increasing the size of the actuator air exhaust piping and solenoid to improve the MSIVBV stroke time.

The NRC staff noted that the PUR and installation of RSGs results in an increase in the post-accident containment temperature and pressure, and requested additional information related to the overpressurization of isolated piping, referencing GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." By letter dated September 6, 2002, the licensee stated that the slight increase in temperature has a minimal effect on equipment in containment, and the existing analyses performed to address GL 96-06 remain bounding for the RSGs and PUR conditions. The licensee concluded that the systems and components within containment subject to post-accident environmental heating and pressurization remain capable of withstanding the predicted peak pressures and temperatures.

The MSIVBVs are discussed further in Section 6.7.

## **6.2 Auxiliary Feedwater System**

The AFW system is designed to supply an independent source of water to the SGs during plant startup, hot standby, normal shutdown, and in the event of loss of the main FW supply. The system consists of two redundant, safety-related essential trains and one nonessential train, all of which supply both SGs. AFW supply is provided by the condensate storage tank (CST) with a backup supply available from the reactor makeup water tank.

The essential AFW trains, which are not routinely used during startup and shutdown, can function automatically as required in the event of a LOP, SBO or an accident. Under these conditions, the decay heat is removed from the SGs to the atmosphere via the ADVs. The major components of the essential AFW trains are two redundant 100 percent capacity safety grade pumps which are powered from redundant and diverse sources (one steam turbine driven pump and one motor driven pump). These two essential AFW trains are cross-connected through redundant normally closed motor-operated valves in series such that either essential AFW pump can supply FW to either or both SGs in the event of a single active failure of the AFW components. A third 100 percent capacity nonessential/ nonsafety grade motor driven pump is provided for use during startup, hot standby, and normal shutdown conditions. This pump may also be manually loaded to an essential emergency bus if necessary.

The licensee re-analyzed the minimum flow requirements of the AFW and the minimum CST inventory dictated by accident analyses of the limiting transients. Results of the re-analyses demonstrate that the AFW performance remains acceptable for PVNGS-2 operations at the proposed 2.94 percent PUR.

The CST, which has a capacity of 550,000 gallons, provides a reserve capacity of 300,000 gallons to support the AFW for secondary side cool-down. The increased decay heat resulting from plant operations at the proposed 2.94 percent PUR will cause a slight increase in the condensate inventory needed for the secondary side cool-down. However, results of the re-analyses also confirm that the existing CST inventory bounds the condensate inventory necessary for PVNGS-2 operations at the 2.94 percent PUR.

Based on our review of the licensee's rationale and evaluation, as set forth above, the NRC staff finds that PVNGS-2 operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the AFW and CST. Also, based on the experience gained from the NRC staff review of PUR applications for similar PWR plants and the results of the licensee's reanalyses that show the AFW and CST performance remaining acceptable for PVNGS-2 operation at the PUR, the NRC staff concludes that the AFW and the CST are acceptable for PVNGS-2 operations at the proposed 2.94 percent PUR.

### **6.3 Condensate and Feedwater System**

The condensate and feedwater system (CFWS) provides FW from the condenser to the SGs. The CFWS includes all components and equipment from the condenser outlet through the containment isolation valves to the SGs and to the heater drain system. The system, with the exception of the portions from the main steam support structure walls for both the downcomer and economizer lines up to the SGs, serves no safety function and is, therefore, designed as nonsafety-related, Quality Group C, and seismic Category I equipment. The portions of the CFWS from the main steam support structure walls for both the downcomer and economizer lines (including containment isolation valves) up to the SGs, are safety related and designed to Quality Group B, seismic Category I requirements in order to assure FW system isolation in accident situations. Adequate isolation is provided at connections between seismic Category I and nonseismic systems, therefore, failure of the nonsafety-related portions of the CFWS will not affect plant safe shutdown.

The licensee evaluated the CFWS for plant operations at the proposed 2.94 percent PUR. The licensee stated that the condensate and FW pump flow rates will increase, resulting from plant operations at the proposed PUR. Condensate flow rates will increase by approximately 1 percent while FW flow rates will increase by approximately 6 percent. However, the CFWS will perform its intended function for PVNGS-2 operations at the proposed 2.94 percent PUR.

Since the CFWS does not perform any safety related function with the exception of containment and FW system isolation in accident situations, and since plant operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the CFWS, the NRC staff did not review the impact of PVNGS-2 operations at the proposed 2.94 percent PUR on the design and performance of the CFWS, except with respect to FW isolation.

The FWIVs, which also serve as containment isolation valves, are designed to isolate the FW system from the SGs during a MSLB, FWLB, or LOCA. Operation of the FWIVs may

cause potentially large dynamic pressure changes and must be considered in the design of the valves and associated piping. The worst case loads occur following a MSLB from no load conditions with both FW pumps in service providing maximum flow following the break. The FWIVs are designed to close against a pressure differential of 1875 psi. The licensee stated that the FW pumps' maximum discharge pressure (deadhead pressure) is 1636 psia, therefore, the existing FWIV design bounds any potential fluid dynamics change associated with plant operations at the proposed 2.94 percent PUR. In addition, the FWIV closure time is not affected by PUR since the resulting PUR differential pressure is bounded by the original system design.

The NRC staff has reviewed the licensee's rationale and evaluation, and determined that the existing FWIV design bounds operation at PUR conditions. On this basis, and in light of the experience gained from the NRC staff review of PUR applications for similar PWR plants, the NRC staff concurs with the licensee that PVNGS-2 operations at the proposed 2.94 percent PUR will have no impact on the FWIVs.

#### **6.4 Circulating Water System**

The circulating water system (CWS) is designed to remove the heat from the condenser to the atmosphere via the mechanical draft cooling tower, thereby maintaining adequately low condenser pressure. The CWS is not required to maintain the reactor in a safe shutdown condition or mitigate the consequences of accidents. The licensee stated that the performance of this system was evaluated for PUR conditions and determined that the system is adequate for PVNGS-2 operations at the proposed 2.94 percent PUR.

Since the CWS does not perform any safety function, and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS-2 operations at the proposed 2.94 percent PUR on the design and performance of the CWS.

#### **6.5 Main Turbine**

The turbine-generator, which converts steam power into electrical power has a turbine control and overspeed protection system. Overspeed protection is accomplished by three independent systems (normal speed governor, mechanical overspeed, and electrical backup overspeed control systems). These overspeed protection systems are designed to satisfy the requirements of GDC 4 with respect to SSCs important to safety being appropriately protected against environmental and dynamic effects, including the effects of missiles, that may result from equipment failure.

The licensee assessed (with the help of the turbine manufacturer) turbine operations with respect to the design acceptance criteria to verify the mechanical integrity of the turbine under the conditions of plant operations at the proposed 2.94 percent PUR. The assessment included a structural evaluation of the turbine components as well as system performance. The existing minimum overspeed setting is 110 percent and maximum overspeed setting is 111 percent. Since plant operations at the proposed power level do not raise the main steam operating pressure above the original design pressure, the turbine manufacturer concluded that the existing overspeed settings remain acceptable. Also, results of the evaluations showed that

there would be no increase in the probability of turbine overspeed. Therefore, the turbine can continue to be operated safely at the proposed 2.94 percent PUR.

Because mainsteam pressure does not increase with the PUR, the NRC staff finds that PVNGS-2 operations at the proposed 2.94 percent PUR do not change the design aspects and operation of the turbine-generator overspeed protection system. Therefore, the turbine-generator overspeed protection system continues to meet the requirements of the GDC 4. For this reason, and based on the experience gained from the NRC staff review of PUR applications for similar PWR plants, the NRC staff agrees with the licensee that operation of the turbine at the proposed 2.94 percent PUR is acceptable.

## **6.6 Main Turbine Auxiliaries**

The turbine auxiliary system, together with the CWS and main condensers, are designed to remove the heat rejected to the condenser by turbine exhaust and other exhausts over the full range of operating loads, thereby maintaining adequately low condenser pressure. The licensee assessed the performance of these systems for the PUR, and determined that these systems are adequate for PVNGS-2 operations at the proposed 2.94 percent PUR.

Since these systems do not perform any safety function and their failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS-2 operations at the proposed 2.94 percent PUR on the design and performance of these systems.

## **6.7 Main Steam Supply System**

The function of the main steam supply system (MSSS) is to deliver steam from the SGs to the high-pressure turbine over a range of flows and pressures covering the entire operating range from system warmup to maximum operating conditions. The system also provides steam to the moisture separator/reheaters, the FW pump turbines, the auxiliary steam system, and the turbine gland seal.

Steam produced in the two SGs is routed by four main steam lines up to the common header at the high pressure turbine. The portions of the MSSS from the SGs up to and including MSIVs, MSIVBs, MSSVs, and ADVs, are located in the flood and tornado protected main steam support structures, and designed as safety-related, Quality Group B and seismic Category I equipment to satisfy the requirements of GDC 2 and GDC 4.

The safety functions of the MSSS are: provision of steam for safety-related auxiliaries and engineered safety feature pumps; provision of a heat sink during certain transients and accidents; limiting RCS pressure during certain transients; SG and MSSS overpressure protection; and termination of MSLB events. Also, the MSSS allows cool-down of the SG via the ADVs when the condenser is not available, as in during an SBO event. The licensee has stated that its evaluation shows systems associated with the MSSS will operate within the same manner as in the current configuration. The PUR will result in an increase in steam/feedwater flow rate; however, these increase values are within the original design specifications for the MSSS. The licensee stated that the increased mass flow rates have been evaluate and found acceptable.

Because the PUR does not create conditions beyond the original design specifications for the MSSS systems, the NRC staff finds that PVNGS-2 operations at the proposed 2.94 percent PUR does not change the design aspects and operations of the MSSS. Therefore, the MSSS satisfies the requirements of GDC 2 and GDC 4.

The main steam lines from the MSIVs to the turbine stop valves and all branch lines are nonsafety-related, and are not required to support safe shutdown of the reactor. These portions of the MSSS are designed to the requirements of American National Standards Institute (ANSI) B31.1, "Power Piping." Since these portions of the MSSS do not perform any safety-related function, and their failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS-2 operations at the proposed 2.94 percent PUR on their design and performance.

The licensee performed an evaluation of the effects resulting from plant operations at the proposed 2.94 percent PUR on the MSSS including the MSIVs, MSIVBVs, MSSVs and ADVs. The licensee stated that the steam flow resulting from plant operations at the proposed 2.94 percent PUR will be  $17.9 \times 10^6$  lbm/hr which is approximately 5.9 percent above the steam flow of  $16.9 \times 10^6$  lbm/hr at the current full power operation. The main steam design conditions of 1270 psig and 575°F remain unchanged and bound all predicted operating conditions for the system and components. The licensee concluded that with the exception of MSIVBVs, plant operations at the proposed 2.94 percent PUR will have an insignificant or no impact on the MSSS and its associated components.

The MSIVBVs allow small amounts of steam to bypass the MSIVs during plant startups. When plant startup conditions require, using the MSIVBVs allows the piping downstream of the MSIVs to heat-up before opening the MSIVs. The MSIVBVs are automatically closed upon receipt of a main steam isolation signal (MSIS). As a result of plant operations at the proposed 2.94 percent PUR, the MSIS setpoint will be raised from >890 psia to >955 psia. The licensee stated that the MSIVBVs will be modified before implementation of this license amendment so that the valves are capable of closing against the increased differential pressure.

Based on our review of the licensee's rationale and evaluation, and the experience gained from our review of PUR applications for similar PWR plants, and because the PUR only affects the MSIVBVs in the MSSS, and as they will be modified to close against the increased differential pressure, the NRC staff concurs with the licensee that PVNGS-2 operations at the proposed 2.94 percent PUR will have an insignificant or no impact on the MSSS.

## **6.8 Miscellaneous Cooling Water Systems**

### **6.8.1 Plant Cooling Water System**

The plant cooling water system (PCWS) is designed to remove heat from the nonsafety-related, normally operating, closed cooling water systems over the full range of normal plant operation. The PCWS uses a portion of the CWS flow from the plant cooling towers to remove heat from the nuclear cooling water system (NCWS), the turbine cooling water system (TCWS), and the condenser vacuum pump seal coolers. Cooled circulating water returned from the cooling towers is pumped in parallel through the TCWS heat exchangers, NCWS heat exchangers and the condenser vacuum pump seal coolers, and is discharged back to the CWS. The licensee

performed an assessment and stated that the PCWS heat loads considered in the original design remain bounding under PUR conditions.

Since the PCWS does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS-2 operations at the proposed 2.94 percent PUR on the design and performance of these systems.

#### **6.8.2 Turbine Cooling Water System**

The TCWS provides cooling for the nonnuclear-related components in the various turbine plant auxiliary systems. The licensee assessed the effects of the increase in heat loads on the TCWS and stated that the design heat loads for the TCWS components remain bounding under PUR conditions.

Since the TCWS does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS-2 operations at the proposed 2.94 percent PUR on the design and performance of this system.

#### **6.8.3 Nuclear Cooling Water System**

The NCWS is designed to supply cooling water to various nonsafety-related components and heat exchangers during plant normal operation including start-up, normal shutdown and hot standby. The NCWS acts as an intermediate barrier between systems that contain or may contain radioactive or potentially radioactive fluids and systems that should not contain such fluids. It serves no safety-related function. The licensee performed evaluations of the effects of the increase in heat loads on the NCWS and stated that the NCWS has the capacity to accommodate the additional heat loads.

Since the NCWS does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS-2 operations at the 2.94 percent PUR on the design and performance of this system.

#### **6.8.4 Essential Cooling Water System**

The essential cooling water system (ECWS) is a closed loop system which serves as an intermediate barrier between the essential spray pond system and systems which contain radioactive or potentially radioactive fluids in order to eliminate the possibility of an uncontrolled release of radioactivity. It provides cooling water to various safety and non-safety systems during all phases of normal plant operation, including startup through cold shutdown and refueling, as well as following an SBO event, LOCA, MSLB or FWLB. The ECWS heat loads resulting from plant operations at the proposed 2.94 percent PUR will increase slightly. The licensee performed assessments to determine the effects of the increases in heat loads on the ECWS.

Based on our review and the experience gained from our review of PUR applications for similar PWR plants, and because the additional heat loads only results in minor temperature increases in the ECWS for normal and accident scenarios, and the design heat loads for the ECWS components still remain bounding under the PUR conditions, the NRC staff finds that plant operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the ECWS and have an insignificant or no impact on the ECWS. Therefore, the NRC staff concludes that the ECWS is acceptable for PVNGS-2 operations at the proposed 2.94 percent PUR.

#### **6.8.5 Spent Fuel Pool Cooling and Cleanup System**

The spent fuel pool (SFP) cooling and cleanup system (SFPCCS) is designed to remove the decay heat generated by spent fuel assemblies stored in the pool, and to clarify and purify the water in the pool. The SFPCCS consists of two independent full-capacity essential SFP cooling trains each with one pump and one heat exchanger, and two separate non-essential cleanup and purification trains each with a fuel pool cleanup pump, filter, strainer, and ion exchanger. Electrical power to each of the SFP cooling pumps is supplied from Class IE emergency power buses. Heat is removed from the SFP heat exchangers by the NCWS. Supplemental SFP cooling can be provided by interconnections to both loops of the shutdown cooling system.

The SFP cooling trains of the SFPCCS are housed in the seismic Category I flood-and-tornado-protected fuel building. The SFPCCS itself, with the exception of the cleanup and purification trains, is designed to Quality Group C, seismic Category I requirements. The SFP cleanup and purification trains can be isolated by manual valves from the SFPCCS. Failure of these nonseismic, Quality Group D cleanup components during a seismic event will not affect operation of the cooling trains.

The SFP cooling system is designed to maintain the SFP water temperature at or below 125°F during normal operation. As stated above, supplemental fuel pool cooling is provided by the shutdown cooling system to maintain the SFP water temperature at or below 145°F during planned full core offload outages or in the event that an unplanned full core offload is performed.

As a result of plant operations at the proposed 2.94 percent PUR, the decay heat load for any specific fuel discharge scenario will increase slightly. The licensee stated that the maximum allowable SFP decay heat load is administratively controlled so that the heat load in the SFP is less than the available SFP cooling train heat removal capability, considering the worst single failure.

In APS's response, dated August 27, 2002, to the NRC staff's RAI, the licensee stated that the SFP thermal analyses do not determine a peak calculated SFP temperature; instead, the thermal analyses utilize the peak allowable SFP temperature as an end point to calculate the maximum allowable heat load. The predicted actual heat load is compared to the maximum allowable heat load to verify that adequate heat removal capability remains. This approach ensures that the SFP cooling train heat removal capacity is greater than the predicted heat load and that the peak SFP temperature will remain lower than the maximum allowable temperature.

The licensee performed an assessment which indicates that the maximum heat load for a full core off load increases from  $45.8 \times 10^6$  Btu/hr to  $47.0 \times 10^6$  Btu/hr. The lowest heat removal

capability (with one SFP cooling train cooled by ECWS and augmented by one shutdown cooling train) during a full core offload is  $49.0 \times 10^6$  Btu/hr.

Also, the SFP has a water temperature monitor system which alarms in the control room when the SFP water temperature reaches 125°F. In the event that the alarm goes off due to high SFP temperature, plant procedures provide direction for the operator to take corrective actions (i.e., place the standby SFP cooling train in operation, place the shutdown cooling system in the SFP cooling assist mode, stop fuel movement, etc.). This will provide additional measures to prevent the SFP water temperature limits from being exceeded.

Based on the review of the licensee's rationale, experience gained from our review of other similar PUR applications, the capacity of SFP and shutdown cooling trains, and the fact that the plant has administrative controls and operating procedures in place to ensure that backup cooling capability is provided for all SFP cooling scenarios, the NRC staff finds that the design and operation of the SFP cooling system for PVNGS-2 operations at the proposed 2.94 percent PUR is acceptable.

Since the SFP cleanup and purification system does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of plant operations at 2.94 percent PUR on the design and performance of this system.

## **6.9 Miscellaneous Mechanical Reviews**

### **6.9.1 Heating, Ventilation, and Air Conditioning Systems**

Heating, ventilation, and air conditioning (HVAC) systems are provided for PVNGS-2 for safety protection, personnel comfort, and equipment protection functions. The HVAC systems are described in UFSAR Sections 6.4 and 9.4.

In a letter dated August 27, 2002, the licensee responded to the NRC staff's RAI of June 14, 2002, to demonstrate that, based on a review of documented design basis calculations, the total heat load increases are within the design margin at PUR conditions, as indicated below:

The impact of PUR on the various HVAC systems is documented in an engineering study prepared in support of the PUR licensing submittal. The auxiliary building ventilation system (ABVS) and control building heating, ventilation, and air conditioning system (CBHVACS) are the only HVAC systems credited in safety analysis that are impacted by operation at PUR conditions. The worst-case event for heat loads on ABVS remains LOCA and MSLB. The worst-case event for heat loads on CBHVACS remains any event resulting in a LOP. The containment heating, ventilation, and air conditioning systems (CHVACS) and turbine building heating, ventilation, and air conditioning system (TBHVACS) systems are not credited in any safety analysis. New individual design calculations were prepared and/or revised as required to evaluate the heat load impact of the RSGs and PUR on the normal operation of the plant HVAC systems.



#### 6.9.1.1 Containment Heating, Ventilation, and Air Conditioning System

The CHVACS include those systems that function during normal plant operation, the containment pre-access period, or during extended shutdown. These systems are not required to operate during any DBA. The HVAC equipment and ducting within the containment and the main steam support structure are designed to retain structural integrity, but is not required to function during and after a SSE. Those portions of CHVACS that penetrate the containment boundary are designed as seismic Category I insofar as they are required to function to maintain containment isolation capability.

The CHVACS normal mode of operation is designed to maintain the temperature and reduce the humidity below 90 percent. The normal cleanup system, together with the normal purge system is designed to control the airborne radioactivity below the level needed for personnel access for inspection, maintenance, and refueling operations. The refueling purge is designed to maintain the airborne radioactivity at a level that permits sustained personnel occupancy during refueling. The containment is maintained at a negative pressure relative to the atmosphere during the purge cycle.

In the event that the concentration of either airborne particulate or iodine activity in the containment is higher than desired levels, air cleaning is accomplished by activating the recirculating filtration unit. This unit is equipped with charcoal and high efficiency particulate air (HEPA) filters to reduce containment airborne radioactivity to acceptable levels. The main steam support structure normal ventilation system provides once through ventilation with 100 percent filtered outside air for the structure.

In a letter dated December 21, 2001, the licensee states that the design basis for the containment heat load calculations considered a RCS with hot leg reactor coolant temperature ( $T_{hot}$ ) of 621°F. This  $T_{hot}$  of 621°F bounds the PUR  $T_{hot}$ , and therefore, the total heat load resulting from PUR will be less than the original design.

By letter dated June 14, 2002, the NRC staff requested additional information from the licensee concerning how, based on a review of design basis calculations, the total heat load increases were determined to be within the design margin at PUR conditions. In a letter dated August 27, 2002, the licensee responded with the following:

- The CHVACS is not credited in any safety analysis.
- The design calculation was performed to compare the normal heat load contribution of the RSGs to that of the OSGs. This calculation considered the increased RSGs' surface area and the improved reflective metal insulation system. Based on the results of this calculation, the RSGs's contribution to the total containment heat load is predicted to be less than that contributed by the OSGs.
- The heat load from insulated piping and components in the containment building is analyzed in an existing design basis calculation for the bulk containment volume. The piping and equipment heat loads considered in this calculation are based on the original plant RCS  $T_{hot}$  of 621°F, which bounds the PUR predicted  $T_{hot}$  (618.9°F with 10 percent tubes plugged) temperatures. Equipment motor heat is based on the motor rated brake

horsepower, and remains bounding for operation at PUR conditions. As a result, the heat loads considered in the existing CHVACS design basis calculation for containment remain conservative with respect to the predicted PUR heat loads.

- The RCS  $T_{hot}$  PUR conditions do not affect the nonsafety-related reactor cavity cooling subsystem and the CEDM cooling units as its original design is based on the original licensed  $T_{hot}$  of 621 °F. In summary, the RSGs and PUR conditions are bounded by the existing CHVACS design.

Since the CHVACS does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of PVNGS-2 operations at the proposed 2.94 percent PUR on the design and performance of the CHVACS.

#### **6.9.1.2 Auxiliary Building Ventilation System**

The normal ABVS is a once-through air system and serves the equipment rooms, access control areas, the mechanical and electrical penetration areas, areas below the 100-ft 0-inch elevation of the main steam support structure, and the remainder of the auxiliary building to maintain an environmental condition suitable for personnel access and limit potential radioactivity release to the atmosphere during normal operation. The ductwork at all levels of the auxiliary building below level 140-ft 0-inch is designed to retain structural integrity, but is not required to function during and after a SSE. During normal plant operation, treated outside air is distributed through the building on a once-through basis.

The Engineered Safety Features (ESF) equipment rooms and the safety-related AFW pump rooms are served by an essential ABVS during emergency operation. The essential system provides individual ESF equipment room cooling and filtered exhaust to the atmosphere during emergency operation. The essential ABVS includes those systems that function post-LOCA within the ESF pump room, and the exhaust system, which maintain the auxiliary building below elevation 100-ft 0-inch at a negative pressure post-LOCA to prevent unfiltered release of possible airborne radioactivity to the surroundings. The ESF pump room and safety-related AFW pump room coolers are designed to maintain the required room temperatures to ensure the operability of the ESF pumps and motors during accident conditions. The ESF equipment room essential air coolers consist of a recirculating air handling unit, including a cooling coil, in each pump room. The essential exhaust filtration system consists of two essential exhaust filtration units shared with the fuel building, and a connecting tunnel and plenum. Following a LOCA, the ESF equipment and safety-related auxiliary FW pump rooms are automatically isolated (at approximately the 100-ft elevation) from the auxiliary building normal HVAC system on receipt of a safety injection actuation signal (SIAS). The building pressure is reduced to a measurable negative pressure relative to below ambient by the fuel building essential exhaust fans for the space below elevation 100-ft 0-inch in the auxiliary building. Air exhausted from the ESF equipment rooms is filtered by the fuel building essential filter units. The essential exhaust filter units are automatically actuated by starting the fans and opening the dampers to the units in response to an ESF pump start signal.

In a letter dated December 21, 2001, the licensee stated the following:

- The increased reactor power level will affect the ABVS. The ABVS system piping design temperatures, pump motor maximum operating horsepower, electrical equipment, lighting heat loads are, with one exception, not affected by the PUR. The increased reactor power does result in an increased post-accident (LOCA and MSLB) containment temperature as discussed in Section 6.2 of the PUR submittal. This affects the transmission of heat loads through the containment wall into the adjacent rooms. However, this increase in heat loads remains bounded by the original ABVS design.
- The heat loads or other input parameters considered in the original SBO design remains bounding for PUR. The essential equipment rooms' original design environmental condition remains bounding for PUR. Heat transfer due to fluid transport through ECCS piping was evaluated. The ABVS heat loads have been revised to account for the new heat loads predicted to occur post-LOCA and MSLB. The heat loads remain within the individual room cooling coil capacities, and within the total ABVS and essential chilled water system capacities.

In a letter dated August 27, 2002, the licensee responded with the following to an NRC staff RAI of June 14, 2002, concerning how total heat load increases are within the design margin at PUR conditions:

- The worst-case event for heat loads on ABVS remains LOCA and MSLB.
- The ABVS system is minimally impacted by the PUR as documented in revisions to the design basis calculations. The increases in post-LOCA and MSLB temperatures have nominally increased the transmission heat loads through the containment wall to the AFW pump and electrical penetration rooms. The increase in temperature remains below the 104°F maximum design temperature.
- The increase in a transmission heat load was evaluated in the design basis for the ABVS and essential chilled water system calculations and remained well within the design capabilities of these systems. The existing ABVS system design basis calculations bound operation at PUR conditions for the piping and equipment heat loads.

The auxiliary building essential ventilation system's and safety-related portions of the auxiliary building normal ventilation system's conformance with the requirements of GDC 2 and GDC 4 to Appendix A of 10 CFR Part 50 is unchanged. Based on the licensee's rationale and evaluation, as set forth above, the NRC staff concurs with the licensee that plant operation at the proposed 2.94 percent PUR will have no impact on the ABVS. In addition, based on the NRC staff experience with reviews of PUR applications at other PWR plants, the NRC staff concludes that plant operation at the proposed 2.94 percent PUR will have no impact on the ABVS.

#### **6.9.1.3 Turbine Building Heating, Ventilation, and Air Conditioning System**

TBHVACS is not a safety-related system. The system operates during normal plant operation and during the shutdown period, depending upon heat removal needs. The TBHVACS includes the turbine building general area ventilation subsystem, switchgear room and battery room ventilation subsystem, and a lube oil room ventilation subsystem. The TBHVACS consists of

supply air handling units and associated equipment to supply air at all levels; approximately 80 percent of this air is exhausted through the roof exhaust fans, and the remaining 20 percent is filtered through wall openings and other exhaust fans.

In a letter dated December 21, 2001, the licensee stated that the TBHVACS heat loads are based upon the piping design temperatures, pump motor rated horsepower, mechanical equipment design temperatures, electrical, control equipment and lighting loads, and transmission loads from adjacent rooms/structures. The heat loads used for the original plant design remains bounding for the PUR heat loads.

In a letter dated August 27, 2002, the licensee responded with the following to the NRC staff RAI of June 14, 2002, concerning how the total heat load increases are within the design margin at PUR conditions:

- The TBHVACS is not credited in any safety analysis.
- The turbine building heat loads evaluated in the design basis calculations remain bounding for operation at PUR conditions. These calculations are based on the original plant piping design temperatures and considered the OSG's higher secondary operating pressure and temperature.

The TBHVACS has no safety function. The system is separated from safety-related plant systems and areas; therefore, failure of the system will not compromise the operation of any essential plant systems or result in an unacceptable release of radioactivity and, therefore, PVNGS-2 continues to conform with the requirements of GDC 2.

Since the TBHVACS is not credited in any safety analysis, and its failure will not affect the performance of any safety-related system or component, the NRC staff did not review the impact of plant operations at the proposed 2.94 percent PUR on the designs and performances of the TBHVACS.

#### **6.9.1.4 Control Building Heating, Ventilation, and Air Conditioning System**

The CBHVACS includes an essential HVAC subsystem and a normal HVAC subsystem. Both HVAC subsystems are provided for the following two areas:

- Control room, computer room, and associated rooms at elevation 140-ft. The essential HVAC system, as well as the habitability systems for the control room, are discussed in Section 6.4.
- ESF switchgear, ESF equipment rooms, and battery rooms without ventilation. Temperatures in the ESF air handling units' room will be less than 96°F. Essential equipment in these areas is qualified to this temperature.

The HVAC system for the upper and lower cable spreading rooms operates in the normal mode only, and is included as a part of the normal HVAC system of the ESF switchgear, ESF equipment rooms, and battery rooms. Without ventilation, temperatures in the upper and lower cable spreading rooms will be less than 105°F. Essential equipment in these areas is qualified to this temperature.

In a letter dated December 21, 2001, the licensee stated:

- The CBHVACS heat loads are based on the piping design temperatures, pump rated motor horsepower, mechanical equipment design temperatures, electrical and control room lighting loads, transmission loads from adjacent structures, and personnel loads. The heat loads used for the original plant design remain bounding for the PUR heat loads.
- Modifications to instruments installed in the control room do not affect the control room heat loads.

In a letter dated August 27, 2002, the licensee responded to the NRC staff's RAI of June 14, 2002, and stated that the control building heat loads evaluated in the CBHVACS design basis calculations are not impacted and remain bounding for operation at PUR conditions. The CBHVACS is credited in safety analyses that are impacted by operation at PUR conditions. The worst-case event for heat loads on CBHVACS remains any event resulting in a LOP.

PVNGS-2's redundancy in emergency systems is unchanged and its compliance with the guidance of RG 1.52, "Design, Inspections, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," and SRP Section 6.4 is continued. The essential system and portions of the normal system are seismic Category I, Quality Group C and thus PVNGS-2 continues to conform with GDC 2. Since the outside air intakes are located in a concrete tornado missile protected plenum, these systems in PVNGS-2 continue to conform with GDC 4.

Based on the licensee's rationale and evaluation, the staff concurs with the licensee that plant operation at the proposed 2.94 percent PUR will have no impact on the CBHVACS. In addition, PVNGS-2 operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the CBHVACS, and based on our experience with reviews of PUR applications at other PWR plants, the NRC staff concludes that plant operation at the proposed 2.94 percent PUR will have no impact on the CBHVACS.

#### **6.10 Low Temperature Overpressure Protection**

Low temperature overpressure protection (LTOP) is a set of actions prescribed to ensure that the vessel will not be subjected to conditions of brittle fracture. This is accomplished using the shutdown cooling system relief valves, whenever the cold leg temperature is below a predetermined "LTOP enable" temperature and there is the possibility of vessel pressurization.

The NRC staff reviewed the low temperature overpressure protection limits to confirm (1) that the vessel fluence calculations were carried out using acceptable analytical methods, and (2) that the results are conservative.

PVNGS-2 has approved 32 effective full-power year (EFPY) pressure temperature curves, therefore, the fluence methodology and the systems calculations have been reviewed and approved. As stated in Section 5.4, the fluence calculation of record is conservative and bounds the value calculated for the proposed PUR. Similarly, the NRC staff concludes that an approved method was used for the transient analysis (i.e., heat and mass input transients).

Regarding the impact of the proposed PUR, the parameters affected are: decay heat, greater RCS and SG secondary volume, greater SG metal mass, and changed SG hydraulic characteristics. The mass and energy addition transients are impacted. The licensee performed calculations which account for the new parameter values and established that the existing pressure temperature curves bound the limits resulting from the parameter changes due to the PUR. The NRC staff finds this conclusion reasonable because there exists a large margin in the estimated fluence and consequently in the material properties at the uprated level. Therefore, the current pressure temperature and LTOP limits are acceptable for operation at the proposed PUR level.

The NRC staff review of the information submitted regarding pressure temperature curves and the LTOP, as set forth above, indicates that the current pressure temperature curves and LTOP limits are applicable for operation at the PUR level.

## **6.11 Miscellaneous Electrical Reviews**

### **6.11.1 Offsite Power System**

In accordance with GDC 17 of 10 CFR Part 50, "Electric Power Systems," the offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covered the information, analyses and documentation for the offsite power system and the stability studies for the electrical transmission grid. The focus of the review relates to the basic requirement that loss of the nuclear unit, the largest operating unit on the grid or the loss of the most critical transmission line will not result in the loss of offsite power to the plant. SRP Branch Technical Position (BTP) ICSB 11, "Stability of Offsite Power Systems," outlines an acceptable approach to addressing the issue of stability of offsite power systems. Acceptance criteria are based on GDC 17, and specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to 8.2 and BTPs PSB-1 and ICSB-11.

Because the licensee evaluated all single-contingency disturbances and determined that grid stability is maintained, the NRC staff concludes that the impact of the PUR on grid stability is insignificant, and the offsite circuits continue to be appropriately designed with respect to grid stability under PUR conditions. Based on this, the staff finds the PUR acceptable in this regard.

The licensee evaluated the impact of the main power transformers (one transformer per phase) and determined that the PUR will raise main transformer oil temperature. This oil temperature increase remains below the transformer's rated temperature capacity, however. Although the PUR analysis does not call for any change, the main transformer will be modified to increase reliability. Accordingly, the staff finds the PUR acceptable in this regard.

The isolated phase bus (isophase bus) is the electrical connection from the main generator output terminals to the low voltage terminals of the main transformer and to the high voltage terminals of the unit auxiliary transformer. The isophase bus rating is 1600 MVA (forced cooling). This rating is greater than the main generator and main transformer ratings, and therefore, the PUR will have no impact on the isophase bus, and the staff finds the PUR acceptable in this regard.

The licensee evaluated the impact of an increased load from the RCPs, heater drain pumps, etc. on the startup transformers/unit auxiliary transformers. The total electrical load increase

due to the PUR is within the rated capacities of the startup transformers/unit auxiliary transformers. The effect of the horsepower load increase on the non-Class 1E 13.8 and 4.16 kV auxiliary electrical distribution system was also evaluated. The affected 13.8 kV switchgear, circuit breakers and cables and 4.16 kV transformers, switchgear, circuit breakers and cables were compared to their rated capacities. The values increased but were below each component's rated capacity. Therefore, the NRC staff concurs that the PUR conditions will have no impact on the startup transformers/unit auxiliary transformers.

#### **6.11.2 AC Power Systems (Onsite)**

The AC onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the ac onsite power system. Acceptance criteria are based on GDC 17 as it relates to the capability of the ac onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

The two EDGs provide an independent source of Class E onsite power (4160 volts) for each of the two trains of ESF. Evaluation of the loads on each of the safety buses demonstrates that those loads do not increase due to the PUR. Therefore, the EDGs remain capable of supplying ESF equipment during all operating and accident conditions. Therefore, the staff finds the licensee's proposed PUR acceptable with respect to the onsite AC power system.

#### **6.11.3 DC Power Systems (Onsite)**

The direct current (DC) power systems include those DC power sources and their distribution systems and auxiliary supporting systems provided to supply motive or control power to safety-related equipment. The staff's review covered the information, analyses, and referenced documents for the DC onsite power system. Acceptance criteria are based on GDC 17 and 10 CFR 50.63, "Loss of all Alternating Current Power," as they relate to the capability of the onsite electrical power to facilitate the functioning of SSCs important to safety. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2.

The DC loading requirements were reviewed, and no reactor power-dependent loads were identified. Operation at the PUR level does not increase any loads or revise control logic. Therefore, there are no changes to the load, voltage drop or short circuit current values. Therefore, the staff finds the licensee's proposed PUR acceptable with respect to the design of the DC onsite power system.

#### **6.11.4 Station Blackout**

The term "station blackout" refers to the complete loss of AC electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO therefore involves the loss of offsite power concurrent with turbine trip and unavailability of the onsite emergency AC power system, but not the loss of available AC power to buses fed by station batteries through inverters or the loss of power from "alternate AC sources" (AAC). The staff's review covered the station blackout events, which are defined for an established period of time, and from which

the plant is required to recover. Acceptance criteria are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and 8.2 Appendix B.

The two SBO gas turbine generators provide an AAC electrical power of sufficient capacity and reliability to operate the systems required for coping with an SBO for a minimum of 4 hours. The PUR will not add any additional load. The methodology and the assumptions associated with the SBO analysis with regard to equipment operability are unchanged with the PUR. Therefore, the ability of the plant to respond to an SBO and to recover therefrom will not be altered due to the PUR, and the staff finds the PUR acceptable in this regard.

#### **6.11.5 Environmental Qualification of Electrical Equipment**

The term "environmental qualification" applies to equipment important to safety to assure this equipment remains functional during and following DBEs. The staff's review covered the environmental conditions that could affect the design and safety functions of electrical equipment including instrumentation and control. The staff's review is to ensure compliance with the acceptance criteria thus ensuring that the equipment continues to be capable of performing its design safety functions under all normal environmental conditions, anticipated operational occurrences, and accident and post accident environmental conditions. Acceptance criteria are based on 10 CFR 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants," as it relates to specific requirements regarding the qualification of electrical equipment located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

In accordance with 10 CFR 50.49, safety related electrical equipment must be qualified to survive the temperature, pressure, and radiation environment at their specific locations during normal and accident operating conditions. The revised LOCA and MSLB analyses resulted in a change to the environmental parameters for the equipment required by 10 CFR 50.49. The licensee reviewed these changes in the environmental parameters for impact on the qualification of electrical equipment, such as radiation, temperature, and pressure. The licensee compared the equipment exposure doses inside and outside the containment building. This comparison demonstrated that the electrical equipment remains qualified as required by 10 CFR 50.49 and in accordance with IEEE 323-1974, since temperature, pressure, and radiation are still bounded by PVNGS-2's original qualification. The staff concludes that the environmental qualification of the electrical equipment is acceptable and meets the relevant requirements of 10 CFR 50.49 with the exception of the In-Core Instrument connectors and non-standard Raychem splices, which will be qualified before implementation of this amendment.

#### **6.12 Essential Spray Pond System**

The ultimate heat sink (UHS) provides heat dissipation capability for the reactor and its essential equipment through the essential spray pond system (ESPS) during normal shutdown, refueling, and accident conditions. The UHS consists of two independent seismic Category I essential spray ponds. Each pond serves one train of the ESPS. Redundant manually operated seismically qualified butterfly valves are provided between ponds, such that the total inventory from both ponds is available to either ESPS train. Warm water returned to the ponds is pumped through ESPS spray nozzles. Heat dissipation is by evaporation to the atmosphere.



The spray nozzles are designed to provide an optimum spray spectrum and are arranged to minimize interference between sprays. Makeup to the ponds is provided by the nonessential domestic water system with a backup source available from the nonessential station makeup water reservoir.

In accordance with the Improved Technical Specifications, the UHS is only required to have sufficient water inventory without makeup for a duration of 26 days following a LOCA. During the operating license application review for PVNGS-2, the staff, in Supplement No. 3 to its SE dated September 1982, concluded that the guidance in RG 1.27 regarding sufficient UHS water capacity was satisfied.

Based on our review of the licensee's rationale and the experience gained from our review of PUR applications for similar PWR plants, we find that plant operations at the proposed 2.94 percent PUR do not change the design aspects and operations of the UHS, and that the PUR has an insignificant or no impact on the UHS and ESPS. Therefore, the staff concludes that the UHS and ESPS remain acceptable for PVNGS-2 operations at the proposed 2.94 percent PUR.

## **7.0 MISCELLANEOUS TOPICS**

### **7.1 Post-Loss-of-Coolant Accident Hydrogen Generation**

APS's PURLR discusses the licensee's review of the effect of the PUR on hydrogen generation and distribution in the containment. The licensee used the analysis methods discussed in the UFSAR to analyze the effects of post-accident hydrogen generation.

The licensee determined the core wide oxidation rate following a LOCA at PUR conditions to be 0.86 percent. This is less than the limit specified in 10 CFR 50.46 and is therefore acceptable. The existing inventories of aluminum and zinc are not changed by the PUR. The hydrogen recombiners are assumed to start at 100 hours. This is consistent with the PVNGS-2 design basis (UFSAR Section 6.2.5.1). The licensee has determined that the peak predicted hydrogen concentration in the containment remains below 3.99% by volume and is therefore acceptable.

Under postulated LOCA conditions, the Reactor Drain Tank room may become essentially a closed room with the venting only from an annular opening in the ceiling. This situation is discussed in UFSAR Section 6.2.5. The licensee's treatment of this situation was previously found acceptable in an NRC SE dated May 23, 1996 (Reference 9-11 of PURLR). The licensee used the same methodology as used for the original design to show that the gas plume leaving the room remains below the combustible limit. This conclusion is based on conservative post-LOCA conditions which bound those predicted for the PUR. The NRC staff finds the licensee's analysis acceptable since: (1) the core oxidation following a LOCA is less than 1 percent, (2) the hydrogen concentration in the main containment volume remains less than 4 percent and the gas plume from the Reactor Drain Tank room is below the combustible limit when analyzed at the higher power, and (3) acceptable methods were used to perform these analyses.

### **7.2 Fire Protection Program**

GDC 3, "Fire protection," addresses generic issues for nuclear power plants regarding the design of SSCs to minimize the probability and effects of fires and explosions. In addition,

10 CFR 50.48 defines requirements for licensees' fire protection programs. Being licensed to operate on April 24, 1986, PVNGS-2 is not subject to the requirements of Appendix R to 10 CFR Part 50, which is only applicable to plants licensed to operate prior to January 1, 1979. The requirements have been reflected in SRP Section 9.5.1, and the approval and requirements for implementing the fire protection program have also been incorporated into an operating license condition for each operating unit. The staff reviewed the licensee's PUR submittals to determine the effects of increasing the plant power rating on the approved fire protection plan.

In Section 9.6 of the PURLR, "Fire Protection Program," the licensee states that operation of PVNGS-2 at the proposed 2.94 percent PUR will not affect the design or operation of the plant's fire detection systems, fire suppression systems, or fire barrier assemblies installed to satisfy NRC fire protection requirements, or result in an increase in the potential for a radiological release resulting from a fire. Any changes to the plant configuration or combustible loading as a result of modifications necessary to implement the PUR will be evaluated by the licensee under the plant's existing NRC-approved fire protection program.

The licensee performed a thermal-hydraulic analysis of the important plant process parameters following a fire assuming PUR conditions. This analysis indicates that only the operator time constraints related to the time needed to deplete the CST and the reactor makeup water tank volumes during plant cool down are affected by the PUR. The licensee has concluded that the safe shutdown methodology and results identified in the UFSAR are maintained considering the modified operator response times for the PUR. All other important plant process parameters and time constraints remain unchanged. The licensee has made no other changes to the plant's hot standby SSCs, components or procedures necessary to achieve and maintain cold shutdown conditions within 72 hours.

The NRC staff has reviewed the licensee's rationale and assessment, and based on that rationale, we conclude that the licensee has adequately considered the effect of the PUR on the fire protection program.

### **7.3 High Energy Line Breaks Outside Containment**

With regard to protection for SSCs important to safety against pipe breaks outside containment, PVNGS-2 is designed in accordance with the guidelines of SRP BTP ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment." The plant design accommodates the effects of postulated pipe breaks and cracks, including pipe whip, jet impingement and environmental effects. The means used to protect essential/safety-related systems and components include physical separation, enclosures, pipe whip restrainers, and equipment shields.

The licensee assessed the systems evaluated in the UFSAR to determine the effects of plant operations at the proposed 2.94 percent PUR on high energy line breaks (HELB) outside containment. System operating parameters for the PUR were assessed against the system pressure and/or temperature parameters used in the existing plant bases to demonstrate the acceptability for HELB effects. The assessment shows that plant operations at the proposed 2.94 percent PUR have an insignificant or no impact on the consequences (e.g. environmental pressure and/or temperature parameters, etc.) resulting from HELB outside containment. In addition, there is no impact on the methods of protection of safety-related systems from HELBs.

We reviewed the licensee's rationale, and based on that rationale, as set forth above, and the experience gained from our review of PUR applications for similar PWR plants, the staff concurs with the licensee's conclusion that PVNGS-2 operations at the proposed 2.94 percent PUR have an insignificant or no impact on the consequences (e.g. environmental pressure and/or temperature parameters, etc.) resulting from HELB outside containment. Therefore the staff concludes that the PUR is acceptable in this regard.

#### **7.4 Erosion/Corrosion Program**

An increase in power will affect fluid velocities, temperatures, and moisture content within many systems of the plant. These parameters directly influence the erosion/corrosion characteristics of these systems. The licensee has established an erosion/corrosion program to monitor pipe wall thinning in single and two-phase flow in high energy carbon steel piping and for identifying corrosion damaged components that should be repaired or replaced. This erosion/corrosion program identifies the piping components and locations that should be monitored for flow accelerated corrosion (FAC). NRC GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," provides guidance for inspecting pipes and components subject to FAC and requests that an effective program be implemented to maintain structural integrity of high-energy carbon steel systems.

In an August 29, 2002, letter, the licensee stated that its FAC program fully conforms to NRC GL 89-08. The plant components which are susceptible to FAC are modeled in CHECWORKS, EPRI's predictive model for FAC. The program and procedures are in place to monitor and maintain the structural integrity of high-energy carbon steel piping. Changes to the model due to the PUR will be made before the installation of the RSGs and implementation of the PUR. The changes will include all parameters affecting FAC and at that time the component wear rates before and after the PUR can be compared. The licensee stated that the model changes are expected to be minor and will result in insignificant changes to plant component susceptibility to FAC.

On the basis of the information the licensee provided, the staff concludes that the proposed PUR is acceptable with respect to the erosion/corrosion program because (1) the PUR will result in negligible effects on the parameters that influence FAC and on the licensee's erosion/corrosion program, and (2) the erosion/corrosion program has adequate provisions to manage erosion/corrosion in high energy piping under the PUR conditions.

#### **7.5 Flooding**

To assure conformance with the requirements of 10 CFR Part 50, Appendix A, GDC 2, "Design bases for protection against natural phenomena," the review of the plant flood protection include all SSCs whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

##### **7.5.1 Containment Sump pH and Containment Flooding**

The licensee re-examined the effect of increased steam generator primary water inventory on the post-LOCA containment flood level. The licensee concluded that the increase in water was small compared to the post-LOCA total volume of water in the containment and does not change the conclusion of the existing analyses that the containment flood level remains below

the current acceptable level. Since the current containment flood limit is not exceeded, the staff finds the PUR to be acceptable with respect to containment flooding.

### **7.5.2 Outside Containment Flooding**

The source of flooding at the site is a probable maximum flood (PMF) for various rivers and washes in site vicinity. These desert water courses are normally dry with flow occurring in them only as a result of rainfall runoff. The PVNGS-2 site is located at an elevation above the PMF level occurring in the desert streams in the area. Therefore, all safety-related systems and components are located above the PMF level.

During the PVNGS-2 operating license application review, the staff, in an SE report dated November 1981, concluded that the design of the facility for flood protection conforms to the requirements of GDC 2 with respect to protection against natural phenomena, and conforms to the guidelines of NRC RG 1.102 concerning flood protection. Also, the plant has adequate protection for safety-related equipment from the effects of postulated piping failure outside containment in accordance with the guidelines of SRP BTP ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

Since events of natural phenomena are not power dependent, we conclude that PVNGS-2 operations at the proposed 2.94 percent PUR will not result in an increase in the probability of flooding caused by PMF.

With respect to building flooding in areas outside the containment, the worst-case flow from high or moderate energy piping systems including the main steam support structure and main FW system are used in the existing flooding analyses. The licensee performed evaluations on the effect of the PUR on building flooding in areas outside the containment and concluded that the effect of the PUR on building flooding in areas outside the containment is bounded by the existing analysis as discussed in the UFSAR Section 3.6.1.

We have reviewed the licensee's rationale, and based on that rationale, as set forth above, and the experience gained from our review of PUR applications for similar PWR plants, the staff concurs with the licensee's conclusion that PVNGS-2 operations at the proposed 2.94 percent PUR will have an insignificant or no impact on building flooding resulting from high or moderate energy pipe break outside containment.

## **7.6 Human Factors Considerations**

The following is the guidance with respect to information on human factors considerations that is needed by the NRC staff as part of its review of proposed power uprates.

### **1. Changes in Emergency and Abnormal Operating Procedures**

Describe how the proposed power uprate will change the plant emergency and abnormal operating procedures.

### **2. Changes to Operator Actions Sensitive to PUR**

Describe any new operator actions needed as a result of the proposed PUR. Describe changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the PUR.

3. Changes to Control Room Controls, Displays and Alarms

Describe any changes the proposed PUR will have on the operator interfaces for control room controls, displays, and alarms. Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed PUR and how operators were tested to determine that they could use the instruments reliably.

4. Changes on the Safety Parameter Display System

Describe any changes the proposed PUR will have on the Safety Parameter Display System. How will the operators know of the changes?

5. Changes to the Operator Training Program and the Control Room Simulator

Describe any changes the proposed PUR will have on the operator training program and the plant referenced control room simulator, and provide the implementation schedule for making the changes.

For Item 1 above, in PURLR Section 9.11.2, the licensee states that an assessment of the expected plant response indicated that minor EOP/Abnormal Operating Procedures (AOP) changes are expected which would not affect operator actions or mitigation strategies that are taken credit for in accident analyses. The licensee explained that the PUR results in changes in operating procedures, such as surveillance tests, normal operating, general operating, and/or alarm response procedures. The procedure changes will be incorporated before operation of Unit 2 at the higher power levels of the PUR.

For Item 2, in PURLR Section 9.11.2, the licensee states that there are no changes to operator credited actions or mitigation strategies.

For Items 3 and 4, in PURLR Section 9.11.1, the licensee states that the PUR will have a limited impact on the operator interfaces for control room displays, controls, and alarms. There will be a few alarm setpoints and indicators changed in the control room because of the PUR. The Qualified Safety Parameter Display System will be modified for the larger SGs that will be installed in the Fall 2003 refueling outage along with the PUR. In all cases, operators are to be trained on the changes before operation at the higher power levels of the PUR. Administrative control procedures provide for such training.

For Item 5, in PURLR Sections 9.11.3 and 9.11.4, the licensee states that because the simulator is modeled after Unit 1, a separate software model will be developed for the Unit 2 changes to plant responses to transient and accident scenarios because of the PUR and larger SGs to support the licensed operator training for Unit 2. The licensee states that this new model will not replace the Unit 1 model for normal examination and evaluation, and that the simulator fidelity will not be affected. The operators for Unit 2 are to be trained on the modifications, Technical Specification changes, procedural changes, and the changes in the Unit 2 response to transients and accident scenarios because of the PUR.

In previous sections of this SE, which are listed below, there have been references to operator actions:

- 4.3.1, "Increase in Heat Removal by Secondary System," states that the NRC review covers operator actions.
- 4.3.1.3, "Steam System Piping Failures Inside and Outside Containment," states that the NRC review covers operator actions.
- 4.3.2.4, "Feedwater System Pipe Breaks," states that the NRC review covers operator actions.
- 4.3.6.2, "Steam Generator Tube Rupture," refers to the operator being able to diagnose the tube rupture and trip the plant manually before reaching the reactor trip point.
- 4.3.6.2.1, "Steam Generator Tube Rupture with Concurrent Loss of Offsite Power (No Stuck Open PORV)," states that the EOPs include explicit instructions to guide the operators to a reactor cool-down.
- 4.4.8, "SGTR with LOP and Single Failure of ADV," states that it is assumed that the operators open the ADVs on both SGs to prevent cycling of the MSSVs and divert FW to the affected SG so as to maintain the SG tubes covered and thereby reduce radioactivity releases.
- 4.4.9, "SGTR with LOP," states that it is assumed that the operators open the ADVs on both SGs to prevent cycling of the MSSVs.

In PURLR Section 9.11.2, the licensee stated that an assessment of the expected plant response indicated that minor EOP/AOP changes are expected which would not affect operator actions or mitigation strategies that are taken credit for in accident analyses. Based on this, the staff concludes that the licensee has properly addressed human factors considerations for the PUR for the proposed amendment.

## **8.0 STATE CONSULTATION**

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the amendment. The State official had no comments.

## **9.0 CONCLUSION**

In summary, the licensee has proposed changes to its operating license and technical specifications to support the replacement of SGs and subsequent operation at an increased maximum power level of 3990 MWt. This proposed PUR would be implemented during plant start-up after the SGs are replaced in PVNGS-2 during refueling outage No. 11, scheduled to begin in September 2003. In this SE, the staff evaluated the analyses submitted by the licensee to verify that results are acceptable and demonstrate compliance to applicable design basis acceptance criteria during the PUR and RSGs conditions.

The NRC staff has concluded, based on the considerations discussed throughout this SE, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in

compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## **10.0 REFERENCES**

1. CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.  
  
CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.  
  
CENPD-133, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.  
  
CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.
2. FLOOD3, Updated version of the NRC approved FLOOD MOD2 computer code.
3. Combustion Engineering Nuclear Power LLC. Topical Report CENPD-140-A, dated June 1976, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis."
4. Bechtel Power Corporation, "Containment Pressure and Temperature Transient Analysis (COPATTA)," as described in Bechtel Topical Report BN-TOP-3, Revision 4, "Performance and Sizing of Dry Pressure Containments."
5. USNRC Bulletin Notice No. 80-04, dated, 2/8/1980, "Analysis of a PWR Main Steam Line Break With Continued Feedwater Addition."
6. USNRC Information Notice No. 84-90, "Main Steam Line Break Effect on Environmental Qualification of Equipment," dated December 7, 1984.
7. NUREG-0588, Interim Staff Position on Environmental Qualification of Safety- Related Electrical Equipment, dated December 1979
8. Combustion Engineering letter DP-456, F. M. Stern to E. Case, dated August 19, 1974, Chapter 6, Appendix 6B to CESSAR System 80 PSAR.
9. ABB-CE Software Verification and Validation Report #00000-AS95-CC-010, Revision 0, Computer code SGNIII, dated December 7, 1995.
10. PCFLUD Computer Program Version 5.0 SQA Classification B, Bechtel Corporation.
11. Letter 102-03578-WLS/AKK/GAM, W.L. Stewart (APS) to Document Control Desk (NRC), "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN-50-528/529/530, Proposed Amendments to Facility Operating Licenses and to Technical Specifications and Various Bases, Related to Power Uprate," January 5, 1996.

12. CENPD-282-P-A, "Technical Manual for the CENTS Code," Volumes 1-3, approved in a letter from M.J. Virgilio, USNRC, to S.A. Toelle, ABB Combustion Engineering, dated March 17, 1994.
13. Letter from C. M. Trammell, USNRC, to W. F. Conway, APS, "Approval of Reload Analysis Methodology Report - Palo Verde Nuclear Generating Station (TAC Nos. M85153, M85154, and M85155)," dated June 14, 1993.
14. Letter from J. Donohew, USNRC, to G. R. Overbeck, APS, "Palo Verde Nuclear Generating Station, Units 1, 2 and 3 - Issuance of Amendments on Peak Fuel Centerline Temperature Safety Limit (TAC Nos. MB6328, MB6329 and MB6330)," dated December 2, 2002.
15. CENPD-135P, "STRIKIN-II - A Cylindrical Geometry Fuel Rod Fuel Rod Heat Transfer Program."
16. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986
17. CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," September 1976.
18. CENPD-207-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Axial Power Distribution," December 1984.
19. CENPD-356(V)-P-A, "Modified Statistical Combination of Uncertainties," Rev. 01-P-A, May 1988.

Attachment:           Table 1  
                          List of Abbreviations

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Date: September 29, 2003



TABLE 1

**NRC STAFF'S ANALYSIS ASSUMPTIONS FOR ACCIDENT DOSE CALCULATIONS****Assumptions Common to One or More Analyses**

Reactor power, 4070 MWt (includes 2 percent uncertainty)

Source Term	Core 4070 MWt <i>Ci</i>	RCS 1% F.F <i>uCi/gm</i>	
Kr-83m	1.69E7	0.013	
Kr85m	5.28E7	1.3	
Kr-85	1.79E6	6.1	
Kr-87	8.77E7	1.0	
Kr-88	1.30E7	2.8	
Kr-89	1.69E8	0.076	
Xe-131m	1.06E6	5.9	
Xe-133m	5.63E6	0.34	
Xe-133	2.29E8	360.0	
Xe-135m	7.39E7	0.74	
Xe-135	2.18E8	7.7	
Xe-137	2.17E8	0.17	
Xe-138	2.02E8	0.63	
I-131	1.02E8	3.0	
I-132	1.55E8	0.83	
I-133	2.29E8	4.4	
I-134	2.68E8	0.52	
I-135	2.08E8	2.5	
RCS mass, lbm			560,000
Initial RCS specific activity, $\mu\text{Ci/gm}$ dose equivalent I-131			
CEA Ejection, LBLOCA, SBLOCA, DBLLOCUS, FSAR Chapter 15.7			3.6
Other			1.0
Initial secondary specific activity, $\mu\text{Ci/gm}$ dose equivalent I-131			0.1
RCS to secondary leak rate @SG, gal/min			0.5
Dose conversion factors			ICRP30 / RG1.109
Offsite breathing rate, $\text{m}^3/\text{sec}$			
0-8 hours			3.47E-4
8-24 hours			1.75E-4
24-720 hours			2.32E-4
Iodine spike appearance rate parameters			
Filtration efficiency fraction			1.0
Letdown flow, gpm			150
RCS initial activity, $\text{uCi/gm}$ d.e. I-131			1.0
RCS leakage, gpm			1.0

Iodine spike duration, hrs	8
Control room volume, ft <sup>3</sup>	1.61E5
Normal ventilation makeup flow, cfm	1200
Essential HVAC system	
Filtered air makeup, cfm	1000
Filtered recirculation, cfm	25740
Unfiltered inleakage, cfm	63
Filter efficiency, elemental, %	95
Filter efficiency, organic, %	95
Filter efficiency, particulate, %	95
Control room breathing rate, m <sup>3</sup> /sec	3.47E-4
Control room occupancy factors	
0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4
Limiting control room $\chi/Q$ (includes occupancy factors), sec/m <sup>3</sup>	
0-8 hrs	1.56E-3
8-24 hrs	1.08E-3
1-4 days	4.15E-4
4-30 days	1.03E-4
Offsite $\chi/Q$ , sec/m <sup>3</sup>	
EAB: 0-2 hr	2.3E-4
LPZ: 0-8 hr	6.4E-5
8-24 hr	4.8E-5
24-96 hr	2.6E-5
96-720 hr	1.1E-5
Essential ESF filtration units efficiency, %	
Elemental	95
Organic	95
Particulate	95

### Assumptions for LBLOCA Analyses

Core release fractions	
Noble gases	1.0
Iodines	0.25

Iodine species fraction		
	<u>Atmosphere</u>	<u>Sump</u>
Particulate/aerosol	0.05	0.0
Elemental	0.91	1.0
Organic	0.04	0.0

Time to CMNT isolation signal (CIAS), sec 12

Control room switchover from normal to emergency mode after CIAS, seconds 50

#### Containment Purge Pathway

Containment purge duration, sec 12

RCS activity released during purge period, % 100

Initial RCS specific activity

    Iodine,  $\mu\text{Ci/gm}$  dose equivalent I-131 60.0

    Noble gases PUR Section 7.6.2 Source term

Containment purge flow rate, cfm

0.00-0.01	39,000
0.01-1.00	35,990
1.00-4:00	37,490
4.00-5.00	33,830
5.00-6.00	26,660
6.00-7.00	19,900
7:00-8:00	13,830
8:00-9:00	8,652
9:00-10:00	4,585
10:00-11:00	1785
11:00-12:00	359

#### Containment Leakage Pathway

Containment volume, $\text{ft}^3$	
Main sprayed region	2.27E6
Auxiliary sprayed region	2.0E5
Unsprayed region	1.5E5
Total	2.62E6

Containment release, %/day

    0-24 hours 0.1

    24-720 hours 0.05

Containment release via depressurized secondary, scfm 0.9

Duration of release, days 30

Containment mixing flow, unsprayed volume change per hour 3.3 (8250 cfm)

Containment air transfer rates, cfm		
Main sprayed to unsprayed		7582
Auxiliary sprayed to sprayed		668
Containment spray lambda, hr <sup>-1</sup>	<u>Main</u>	<u>Aux</u>
Elemental	19.6	6.05
Organic	0.0	0.0
Particulate	0.32	0.09
Containment spray DF		
Elemental iodine		6.51
Plateout		93.4
Containment elemental iodine plateout lambda, hr <sup>-1</sup>		
Main sprayed region		2.14
Auxiliary sprayed region		14.4
Unsprayed region		14.4
Containment spray timings		
Injection spray initiation, sec		92
Injection spray duration, sec		386
Particulate spray reduction, sec		478
Elemental spray cutoff, sec		478

#### ECCS Leakage Pathway

ECCS leak rate, ml/hr	3000
Start of ECCS leakage, minutes	20
Duration of release, days	30
Containment sump volume, ft <sup>3</sup>	7.0E4
Fraction of core iodine inventory in sump	0.5
Iodine flash fraction	0.1

#### Refueling Water Tank Backleakage Pathway

RWT volume, scf	1.15E5
Fuel building volume, scf	7.45E5
Maximum backleakage from SI to RWT, gpm	43
Partition coefficient of iodine in sump water backleakage	1000

Fraction core inventory in gap	0.1
Fraction of gap released	1.0

Time to CIAS signal, sec	129*
Control room switchover from normal to emergency mode after CIAS, seconds	50

Source term	60 uCi/gm Dose Equivalent I-131 Noble gases from table above
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Auxiliary sprayed to sprayed 668

Containment spray lambda, hr <sup>-1</sup>	<u>Main</u>	<u>Aux</u>
Elemental	19.6	6.05
Organic	0.0	0.0
Particulate	0.32	0.09
Containment elemental iodine plateout lambda, hr <sup>-1</sup>		
Main sprayed region		2.14
Auxiliary sprayed region		14.4
Unsprayed region		14.4
Containment spray timings (CSAS=1087 sec)		
Injection spray initiation, sec	33+CSAS	
Particulate spray duration, sec		386
Elemental spray duration, sec		386
<u>Primary-to-Secondary Leakage Pathway</u>		
Primary-to-secondary leak rate @SG, gpm		0.5
RCS source term	100% gap activity plus Initial RCS activity	
Release duration, hours		3
Steam generator partition coefficient		0.01
<u>Initial SG Release Activity Pathway</u>		
Secondary source term, µCi/gm dose equivalent I-131		0.1
Total mass release via MSSVs and ADVs, lbm		334,000
Steam generator partition coefficient		1.0
<u>ECCS Leakage Pathway</u>		
ECCS leak rate, ml/hr		3000
Start of ECCS leakage, minutes		20
Duration of release, days		30
Containment sump volume, ft <sup>3</sup>		7.0E4
Fraction of RCS inventory in sump		0.5
Iodine flash fraction		0.1
Essential ESF Filtration Units Efficiency, %		
Elemental		95
Organic		95
Particulate		95

### **Assumptions for Control Element Assembly Ejection Accident Analyses**

Time to CIAS signal, sec	69
Control room switchover from normal to emergency mode after CIAS, seconds	50

#### Containment Purge Pathway

Containment purge duration, sec	77
RCS activity released during purge period, %	100
Initial RCS specific activity	See table above

#### Containment Leakage Pathway

Source term	
Radial peaking factor	1.77
Fraction of rods that exceed DNB	0.17
Gap fraction, all nuclide groups	0.10
Fraction of rods that exceed DNB that experience melt	0.0
Containment volume, ft <sup>3</sup>	2.62E6
Containment release, %/day	
0-24 hours	0.1
24-720 hours	0.05
Duration of release, days	30

#### Primary-to-Secondary Leakage Pathway

Primary-to-secondary leak rate @SG, gpm	0.5
Steam release via MSSV, ADVs, lbm	1,176,800
Release duration, hours	2.67
Steam generator partition coefficient	0.01

#### ECCS Leakage Pathway

ECCS leak rate, ml/hr	3000
Start of ECCS leakage, minutes	20
Duration of release, days	30
Containment sump volume, gal	522,000
Fraction of RCS inventory in sump	0.5
Iodine flash fraction	0.1

## Assumptions for SGTR with LOP and Single Failure Analyses

(\*\* data interpreted from licensee-provided graphs)

Initial RCS activity	1.0 $\mu\text{Ci/gm}$ dose equivalent I-131	
Initial secondary activity	0.1 $\mu\text{Ci/gm}$ dose equivalent I-131	
Pre-incident iodine spike activity	60.0 $\mu\text{Ci/gm}$ dose equivalent I-131	
Co-incident spike multiplier	500	
Iodine appearance rates, Ci/hr		
I-131		13624
I-132		12005
I-133		24341
I-134		16000
I-135		19790
Iodine spike duration, hrs	8	
Event timing, sec		
Reactor trip		100
LOP		103
Time to SIAS signal, sec	245	
Control room switchover from normal to emergency mode after CIAS, seconds	50	
Break flow flash fraction**		
0-2400 sec		1.0
2400 sec-8 hrs		0.05
Break flow to affected SG**, lbm/sec		
0-60 s		60
60-360 s		46.5
360-1080 s		53.5
1080-3000 s		63
3000-4200 s		57
4200-5760 s		48
5760-7200 s		40.5
7200-12000 s		36
12000-26400 s		31.5
26400-28800 s		30
Primary-to-secondary leakage to unaffected SG, gpm	1.0	
Steam generator mass** @, lbm	Affected	Unaffected
0-60 s	100,000	100,000
60-360 s	70,000	100,000
360-1080 s	55,000	128,000
1080-3000 s	170,000	185,000
3000-4200 s	300,000	262,000
4200-28800 s	300,000	303,000
Steam release from ruptured SG <sup>2</sup> , lbm		
0-2 hours		550,000
2-8 hours		775,000



Steam release from unaffected SGs**, lbm	
0-2 hours	25000
2-8 hours	50000
Steam partition coefficient	0.01
Main condenser DF (prior to LOP)	100

### Assumptions for SGTR with LOP

Initial RCS activity	1.0 $\mu\text{Ci/gm}$ dose equivalent I-131
Initial secondary activity	0.1 $\mu\text{Ci/gm}$ dose equivalent I-131
Pre-incident iodine spike activity	60.0 $\mu\text{Ci/gm}$ dose equivalent I-131
Co-incident spike multiplier	500
Iodine appearance rates, Ci/hr	
I-131	13624
I-132	12005
I-133	24341
I-134	16000
I-135	19790
Iodine spike duration, hrs	8
Event timing, sec	
Reactor trip	760
LOP	763
Time to SIAS signal, sec	780
Control room switchover from normal to emergency mode after CIAS, seconds	50
Break flow flash fraction**	
0-740 s	0.11
740-2700 s	0.035
2700-3000 s	0
3000-3600 s	0.027
3600-5400 s	0.025
5400-6600 s	0.01
6600 s - end	0.00
Break flow to affected SG**, lbm/sec	
0-50 s	60
50-400 s	49.5
400-740 s	45.7
740-1000 s	36
1000-1200 s	40.5
1200-2700 s	42
2700-3000 s	42
3000-3600 s	43.5
3600-5400 s	47.5
5400-6600 s	42.0

6600-7200 s		33
7200-14400 s		15
14400-28800 s		7.5
Primary-to-secondary leakage to unaffected SG, gpm		1.0
Steam generator mass** @, lbm	Affected	Unaffected
0-50 s	115,000	110,000
50-400 s	125,000	110,000
400-740 s	145,000	110,000
740-1000 s	125,000	85,000
1000-1200 s	125,000	85,000
1200-2700 s	185,000	125,000
2700-3000 s	280,000	165,000
3000-3600 s	315,000	210,000
3600-5400 s	330,000	300,000
5400-6600 s	350,000	320,000
6600-7200 s	380,000	300,000
7200-14400 s	450,000	300,000
14400-28800 s	570,000	300,000
Steam release from ruptured SG**, lbm		
0-90 minutes		135,000
Steam release from unaffected SGs**, lbm		
0-2 hours		160,000
2-8 hours		1,015,000
Steam partition coefficient		0.01
Main condenser DF (prior to LOP)		100

#### **Assumptions for Inadvertent ADV Opening Analysis**

Initial RCS activity	1.0 $\mu\text{Ci/gm}$ dose equivalent I-131
Initial secondary activity	0.1 $\mu\text{Ci/gm}$ dose equivalent I-131
Fuel clad damage fraction	0.055
Gap fraction inventory, all nuclides	0.1
Release duration, sec	
Affected SG	1800
Unaffected SG	28800
Primary-to-secondary leak rate to @SG, gpm	0.5
Release holdup	
Affected SG	none
Unaffected SG	yes
Steam partition coefficient	
Affected SG	1.0
Unaffected SG	0.01
Steam generator mass @, lbm	180,000

Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
2-8 hours	2,550,000
Steam release from affected SG (0-30min), lbm	180,000

### Assumptions for MSLB Analyses

Initial RCS activity (1.0% F.F)	3.6 $\mu\text{Ci/gm}$ dose equivalent I-131
Initial secondary activity	0.1 $\mu\text{Ci/gm}$ dose equivalent I-131
Pre-incident iodine spike activity	60.0 $\mu\text{Ci/gm}$ dose equivalent I-131
Co-incident spike multiplier	500
Iodine appearance rates, Ci/hr	
I-131	13624
I-132	12005
I-133	24341
I-134	16000
I-135	19790
Iodine spike duration, hrs	8
Faulted SG blowdown (100%) duration, minutes	30
Primary-to-secondary leakage @SG, gpm	0.5
Primary to secondary leakage duration, days	30
Steam generator mass @, lbm	
Affected	300,000
Unaffected	180,000
Steam release from faulted SG, lbm	300,000
Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
2-8 hours	2,550,000
Steam partition coefficient	
Affected SG	1.0
Unaffected SG	0.01
Time to SIAS signal, sec	49
Control room switchover from normal to emergency mode after CIAS, seconds	50

### **Assumptions for FWLB Analyses**

Initial RCS activity	1.0 $\mu\text{Ci/gm}$ dose equivalent I-131
Initial secondary activity	0.1 $\mu\text{Ci/gm}$ dose equivalent I-131
Primary-to-secondary leakage @SG, gpm	0.5
Primary to secondary leakage duration, hrs	8
Steam generator mass @, lbm	
Affected	300,000
Unaffected	180,000
Steam release from faulted SG (8-hours), lbm	300,000
Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
2-8 hours	2,550,000
Steam partition coefficient	
Affected SG	1.0
Unaffected SG <30 minutes	1.0
Unaffected SG >30 minutes	0.01
Time to SIAS signal, sec	28.4
Control room switchover from normal to emergency mode after CIAS, seconds	50

### **Assumptions for Sheared RCP Shaft Analysis**

Initial RCS activity	1.0 $\mu\text{Ci/gm}$ dose equivalent I-131
Initial secondary activity	0.1 $\mu\text{Ci/gm}$ dose equivalent I-131
Fuel clad damage fraction	0.17
Peaking Factor	1.72
Gap fraction inventory, all nuclides	0.1
RCS mass, lbm	560,000
Primary-to-secondary leak rate to @SG, gpm	0.5
Release duration, hrs	8
ADV sticks open on affected SG, sec	1800
Time to restore affected SG level, sec	5400
Steam partition coefficient	
Affected SG (<90 minutes)	1.0
Affected SG (>90 minutes)	0.01
Unaffected SG	0.01
Steam generator mass @, lbm	
Affected	300,000
Unaffected	180,000

Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
0-8 hours	2,550,000
Steam release from affected SG(0-90 min), lbm	180,000
Time to SIAS signal, sec	351
Control room switchover from normal to emergency mode after CIAS, seconds	50

### **Assumptions for AOO Analysis**

Initial RCS activity	1.0 $\mu\text{Ci/gm}$ dose equivalent I-131
Initial secondary activity	0.1 $\mu\text{Ci/gm}$ dose equivalent I-131
Fuel clad damage fraction	0.10
Peaking Factor	1.72
Gap fraction inventory, all nuclides	0.1
RCS mass, lbm	560,000
Primary-to-secondary leak rate to @SG, gpm	0.5
Release duration, hrs	8
Steam partition coefficient	0.01
Steam generator mass @, lbm	180,000
Steam release from unaffected SGs, lbm	
0-2 hours	1,000,000
0-8 hours	2,550,000

## ACRONYMS

ACRONYM	DEFINITION
AAC	Alternative Alternating Current
ABVS	Auxiliary Building Ventilation System
AC	Alternating Current
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
ANO	Arkansas Nuclear One
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOP	Abnormal Operating procedure
AOR	Analysis of Record
APS	Arizona Public Service
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BSAP	Bechtel Structural Analysis Program
BTP	Branch Technical Position
CBHVACS	Control Building Heating, Ventilation, and Air Conditioning System
CE	Combustion Engineering
CEA	Control Element Assembly
CEAW	Control Element Assembly Withdrawal
CEDM	Control Element Drive Mechanism
CENTS	Combustion Engineering Nuclear Transient Simulator
CEOG	Combustion Engineering Owners Group
CESSAR	Combustion Engineering Standard Safety Analysis Report
CFR	Code of Federal Regulations
CFWS	Condensate and Feedwater System
CFWS	Condensate and Feedwater System
CHF	Critical Heat Flux
CHRS	Containment Heat Removal System
CHVACS	Containment Heating, Ventilation, and Air Conditioning Systems
CPC	Core Protection Computer
CSS	Containment Spray System
CST	Condensate Storage Tank
CUF	Cumulative Usage Factor
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
DBA	Design Basis Accident
DBE	Design Basis Event
DBLLOCUS	Double-Ended Break of the Letdown Line Outside Containment Upstream of the letdown line control valve
DC	Direct current
DEDLSB	Double-Ended Discharge Leg Slot Break
DEHLSB	Double-Ended Hot Leg Slot Break
DESLSB	Double-Ended Suction Leg Slot Break
DF	Decontamination Factor
DNB	Departure from Nucleate Boiling

DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
ECWS	Essential Cooling Water System
EDG	Emergency Diesel Generator
EFPY	Effective Full Power Year
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EQ	Equipment Qualification
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
ESPS	Essential Spray Pond System
FAC	Flow Accelerated Corrosion
FHA	Fuel Handling Accidents
FIV	Flow-induced Vibration
FTC	Fuel Temperature Coefficient
FW	Feedwater
FWIV	Feedwater Isolation Valve
FWLB	Feedwater Line Break
GDC	General Design Criterion
GL	Generic Letter
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HJTC	Heated Junction Thermocouple
HLR	Head Lift Rig
HPPT	High Pressurizer Pressure Trip
HPSI	High Pressure Safety Injection
HVAC	Heating Ventilation and Air Conditioning
I&C	Instrumentation & Controls
ICI	In-Core Instrumentation
ID	Inadvertent Deboration
IOSGADV	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve
LBB	Leak Before Break
LBLOCA	Large Break Loss-of-Coolant Accident
LHGR	Linear Heat Generation Rate
LHR	Linear Heat Rate
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater
LOP	Loss of Offsite Power
LPSI	Low Pressure Safety Injection
LPZ	Low Population Zone
LTC	Long-term Cooling
LTOP	Low Temperature Over Pressure
M&E	Mass and Energy
MCL	Main Coolant Loop
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MSIS	Main Steam Isolation Signal
MSIV	Main Steam Isolation Valve
MSIVBV	Main Steam Isolation Valve Bypass Valve
MSLB	Main Steam Line Break
MSSS	Main Steam Supply System

MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
MWt	Megawatts Thermal
NCWS	Nuclear Cooling Water System
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operational Basis Earthquake
OSG	Original Steam Generator
P <sub>a</sub>	Peak Calculated Containment Internal Pressure for the Design Basis
	Loss of Coolant Accident
PCT	Peak Cladding Temperature
PCWS	Plant Cooling Water System
PLHR	Peak Linear Heat Rate
PMF	Probable Maximum Flood
PPS	Plant Protection System
PRA	Probabilistic Risk Assessment
PSV	Pressurizer Safety Valve
PTS	Pressurized Thermal Shock
PUR	Power Uprate
PURLR	Power Uprate Licensing Report
PVNGS	Palo Verde Nuclear Generating Station
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
ROPM	Required Over Power Margin
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSGs	Replacement Steam Generator
RTP	Rated Thermal Power
RT <sub>NDT</sub>	Nil Ductility Temperature
RVI	Reactor Vessel Internals
RVID	Reactor Vessel Integrity Database
SAFDL	Specified Acceptable Fuel Design Limit
SBCS	Steam Bypass Control System
SBLOCA	Small Break Loss-of-Coolant Accident
SBO	Station Blackout
scfm	Standard Cubic Feet per Minute
SCU	Statistical Combination of Uncertainties
SE	Safety Evaluation
SFP	Spent Fuel Pool
SFPPCS	Spent Fuel Pool Cooling and Cleanup System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SIT	Safety Injection Tank
SRP	Standard Review Plan



SSC	Structure, System, and Component
SSE	Safe Shutdown Earthquake
TAV	Turbine Admission Valve
$T_{ave}$	average reactor coolant temperature
$T_{hot}$	hot leg reactor coolant temperature
TBHVACS	Turbine Building Heating, Ventilation, and Air Conditioning System
TCWS	Turbine Cooling Water System
TS	Technical Specifications
TBV	Turbine Bypass Valve
UFSAR	Updated Final Safety Analysis Report
USE	Upper Shelf Energy
UHS	Ultimate Heat Sink
VOPT	Variable Over-Power Trip